Oak Ridge Associated Universities
Post Office Box 117
Oak Ridge, Tennessee 37831-0117

Energy/Environment Systems Division

June 18, 1991

Mr. Dave Fauver
Division of Low-Level Waste Management and Decommissioning
Office of Nuclear Material Safety and Safeguards
U. S. Nuclear Regulatory Commission
1 White Flint North
11555 Rockville Pike
Rockville, MD 20852

SUBJECT: DOCKET FILE REVIEWS

Dear Mr. Fauver:

Enclosed are two copies of a report, summarizing the findings of ORAU's review of 59 Docket Files for Terminated Research and Test Reactors. The report consists of:

a. Summary capsulations of Docket File contents.
b. A tabulation of facility radiological status, based on Docket contents.
c. A list of decommissioning records not included in the Docket File.

Based on this review it was ORAU's evaluation that 13 files contain sufficient information to conclude that the current status of the facility satisfies the present guidelines for use without radiological restrictions. Residual radioactive material at 16 of the facilities was transferred to another license (AEC/NRC or state) or to an AEC contract; termination of that subsequent license or contract should effectively address the final radiological conditions. It is ORAU's opinion that the remaining 30 files did not contain sufficient information to determine whether the current status meets guidelines or exceeds guidelines. The potential for residual activity above guidelines is, however, considered low for each of these facilities. Review of additional survey information (not available in the Docket File) and/or performance of a few confirmatory measurements would likely be adequate to demonstrate that most of these facilities satisfy the current guidelines for unrestricted use.

If you have questions or comments regarding this information, I may be contacted at FTS 626-3305.

Sincerely,

James D. Berger

James D. Berger, Director
Environmental Survey and
Site Assessment Program

JB:pb

Enclosures

cc: File 169

NLD'|

SUMMARY OF DECOMMISSIONING RECORDS NOT INCLUDED IN DOCKET

Docket File	Record
50-1*	Final site surveys by licensee
50-6*	Final site surveys by licensee
50-8	Final site surveys by licensee
50-13 (CX-10)	Final site surveys by licensee NRC (ORAU) confirmatory survey (same as CX-12 document)
50-13 (CX-12)	NRC (ORAU) confirmatory survey (same as CX-10 document)
50-13/50-191 (CX-1,CX-19)*	NRC confirmatory survey
50-14	Inspection report of March 4, 1970, referenced in May 11, 1970 NRC Memorandum to Docket File
50-24	Final site surveys by licensee Inspection report of October 17, 1969, referenced in November 3, 1969 licensee letter.
50-34*	Final site surveys by licensee NRC confirmatory survey
50-37*	Dismantling and decommissioning plan Final site surveys by licensee NRC confirmatory survey
50-50	Dismantling and decommissioning plan Order authorizing dismantling and disposition of component parts Final site surveys by licensee NRC confirmatory survey
50-60	Order authorizing dismantling and disposition of component parts
50-64	Final site surveys by licensee
50-75	NRC confirmatory survey
50-87(R-119)	NRC confirmatory survey
50-94	Referenced licensee letter of March 26, 1976, supplementing application to terminate licensee
50-101	ORAU report (ORAU 88/G-89) of 1986 site survey
50-106	Final site surveys by licensee (file contains only summary information)

SUMMARY OF DECOMMISSIONING RECORDS NOT INCLUDED IN DOCKET

Docket File	Record
50-141	January 30, 1974 Safety Evaluation Report, referenced in dismantling order. Survey data for soils beneath demolished facility
50-153*	Final site surveys by licensee NRC confirmatory survey
50-172*	Final site surveys by licensee NRC confirmatory survey
50-197	NRC confirmatory survey
50-202	NRC confirmatory survey
50-212*	Final site surveys by licensee NRC confirmatory survey
50-235	Order authorizing dismantling and disposition of component parts
50-246*	Order authorizing dismantling and disposition of component parts Final site surveys by licensee NRC confirmatory survey
50-290	Region I Inspection report 50-290/74-02 December 20, 1973 Safety Evaluation Report referenced in dismantling order ORAU report (ORAU 88/G-89) of 1986 site survey
50-310*	Order authorizing dismantling and disposition of component parts
50-375	ORAU Report of 1986 Confirmatory Survey

* Indicates residual activity or entire facility transferred to another license or AEC contract.

MINIMUM DECOMMISSIONING RECORDS TO EVALUATE
FACILITY RADIOLOGICAL STATUS (TEST/RESEARCH R_x)

1. Applicant Correspondance (incoming)

    a. Application/request to dismantle/decommission facility and terminate license

    b. dismantling and decommissioning plan

    c. final site surveys by licensee

    d. supplemental documentation, if referenced by date, in:

        1. order authorizing dismantling and disposition of component parts

        2. order terminating facility license

2. Applicant Correspondance (outgoing)

    a. environmental assessment and safety evaluation for order authorizing dismantling of the reactor and disposition of component parts

    b. order authorizing dismantling of facility and disposition of component parts

    c. environmental assessment for termination of facility license

    d. order terminating facility license

3. Inspection Reports

    a. NRC confirmatory survey

DOCKET FILE REVIEW RADIOLOGICAL STATUS SUMMARY

Docket/License	Decom. Records Complete	Survey Results[a]					Facility Radiological Status[d]	Remarks
		Exposure Rate[b]	Surface Contam.[b]	Removable Contam.[b]	Other	Survey Documents[c]		
50-538/R-127	Yes	MG	MG	MG		complete	MG	
50-1/R-3	No	DL	NR	NR		MP, SE, SR GI	LT	
50-4/R-005	Yes	DL	NR	MG		MP, SE, SR	LT	
50-6/R-4	No	DL	NR	MG		MP, SE, SR, CI, GI	LT	
50-13 & 191/ CX-1, CX-19	No	MG	II	II		MP, SE, SR, CI, GI	LT	
50-17/R-46	Yes	DL	MG	MG		complete	LT	
50-23/CX-8	*	*	*	*		*	LT	*Facility transfer.ed intact to another license.
50-34/CX-6	No	NR	NR	NR		MP, SE, SR, CI	LT	
50-37/CX-3	No	NR	NR	NR		MP, SE, SR, CI, GI	LT	
50-87/CX-1	Yes	II	NR	II, UC		MP, SE, SR, CI	LT	
50-153/CX-16	*	*	*	*		*	LT	*Facility transferred to AEC contract.
50-167/R-68	No	NR	NR	NR		some provided	LT	
50-172/R-86	No	NR	NR	NR		MP, SE, SR, CI	LT	

[a]Includes results from licensee and confirmatory surveys

[b]MG meets unrestricted release guidelines
EG exceeds unrestricted release guidelines
NR not reported in file
DL detection limit above guidelines
II insufficient information
UC units not compatible with guidelines

[c]MP no maps of survey locations
SE survey equipment and/or techniques not described
IS some facility areas not surveyed
CI contaminants not identified
SR specific results not provided
GI guidelines not identified

[d]MG meets current guidelines
LP low contamination potential
LT license transferred

(6-17-91)

DOCKET FILE REVIEW RADIOLOGICAL STATUS SUMMARY

Docket/License	Decom. Records Complete	Survey Results[a]					Facility Radiological Status[d]	Remarks
		Exposure Rate[b]	Surface Contam.[b]	Removable Contam.[b]	Other	Survey Documents[c]		
50-13/CX-10	No	MG	MG	MG		*	MG	*no licensee survey data; confirmatory survey documents complete.
50-13/CX-12	No	MG	MG	MG		*	MG*	*status based on confirmatory survey of CX-10 facility.
50-87/R-119	No	MG	MG	MG		*	MG*	*status based on confirmatory survey report.
50-101/R-49	No	MG	MG	MG		complete*	MG	*include copy of reports of ORAU 1986 survey.
50-106/R-51	No*	MG	MG	MG		SR*	MG	*file contains summary of licensee's survey; include copy of licensee's data.
50-112/R-53	Yes	MG	MG	MG		MP	MG	
50-124/R-62	Yes	MG	MG	MG		complete	MG	
50-290/CX-25	No	MG*	MG*	MG*		complete*	MG*	*based on 1986 ORAU survey (ORAU-88/G-89)--include copy of report in Docket File.
50-294/R-114	Yes	MG	MG	MG		complete	MG	
50-375/R-118	Yes	MG*	MG*	MG*		complete*	MG*	*based on 1986 ORAU confirmatory survey -- include copy of report in Docket File.
50-394/R-121	Yes	MG	MG	MG		complete	MG	
50-433/R-124	Yes	MG*	MG*	MG*		complete*	MG*	*based on 1989 confirmatory survey by ORAU

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Docket/License	Decom. Records Complete[e]	Survey Results[a]					Facility Radiological Status[d]	Remarks
		Exposure Rate[b]	Surface Contam.[b]	Removable Contam.[b]	Other	Survey Documents[c]		
50-212/R-96	No	NR	NR	NR		some provided	LT	
50-227/R-100	Yes	NR	II, UC	MG		MP, SE, SR, CI	LT	
50-246/CX-24	No	NR	NR	NR		some provided	LT	
50-310/R-72	No	NR	NR	NR		MP, SE, SR, GI	LT	
50-8/R-1	No	II	NR	UC		MP, SE, SR, CI, GI	LP	
50-14/CX-9	No	NR	NR	MG		MP, SE, SR, CI, GI	LP	
50-24/CX-4	No	NR	II	II		MP, SE, SR, CI, GI	LP	
50-43/R-11	Yes	DL	NR	II, UC		MP, SE, SR, CI, GI	LP	
50-50/R-19	No	NR	NR	NR		MP, SE, SR, CI, GI	LP	
50-58/R-22	Yes	NR	NR	II		MP, SE, SR, IS, GI	LP	
50-60/R-27	No	MG	NR	MG		SE, SR, CI, GI	LP	
50-64/R-24	No	NR	NR	II		MP, SE, SR, CI, GI	LP	

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		Exposure Rate[b]	Surface Contam.[b]	Removable Contam.[b]	Other	Survey Documents[c]		
50-75/CX-13	No	NR	MG	MG		SR, CI	LP	
50-84/R-30	Yes	EG*	II, UC	II, UC		MP, SE, SR, CI, GI	LP	*data on reactor components only.
50-94/R-40	No	II	II	MG		MP, SE, SR, CI	LP	
50-98/R-43	Yes	DL	MG	MG		complete	LP	
50-99/R-47	Yes	MG	NR	MG		MP, SE, IS	LP	
50-108/CX-15	Yes	DL	II	II		MP, SE, SR, CI	LP	
50-111/R-63	Yes	MG	NR	MG		SE	LP	
50-114/R-54	Yes	DL	II, UC	MG		MP, SE, SR, CI	LP	
50-122/R-55	Yes	DL	NR	MG		CI, GI	LP	
50-129/R-58	Yes	MG	II	MG		MP, SE, IS, CI	LP	
50-135/R-85	Yes	DL	NR	MG		MP, SE, SR, CI	LP	
50-141/R-60	No*	EG	EG	MG	soil samples-NR	SE, SR*	LP	*confirmation of floor demolition and follow-up soil and exposure rate measurements should be included in file.
50-147/CX-17	Yes	MG	UC	MG		MP, SR	LP	

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DOCKET FILE REVIEW RADIOLOGICAL STATUS SUMMARY

Docket/License	Decom. Records Complete	Survey Results[a]					Facility Radiological Status[d]	Remarks
		Exposure Rate[b]	Surface Contam.[b]	Removable Contam.[b]	Other	Survey Documents[c]		
50-154/CX-18	Yes	II	II, UC	II, UC		MP, SE, SR, CI, GI	LP	
50-197/CX-21	No	NR	MG	MG		SE, IS, CI	LP	
50-202/R-91	No	NR	NR	MG*		SE, IS, CI	LP	*exception is vault which was later demolished; no final survey data provided after vault demolition.
50-203/CX-20	Yes	II	II	II		MP, SE, SR, CI	LP	
50-234/CX-23	Yes	NR	II, UC	MG		MP, IS	LP	
50-216/R-107	Yes	NR	II, UC	MG		MP, SR, CI	LP	
50-235/R-99	No	NR	II, UC	MG		MP, SE, SR, CI, GI	LP	
50-240/R-104	Yes	DL	II, UC	MG		MP, SE, SR, CI	LP	
50-253/R-105	Yes	DL	II, UC	MG		MP, SE, SR, CI	LP	

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EXPLANATION OF INFORMATION OR STATUS CODES

Survey Results

1. Exposure rate/Surface Contamination/Removable Contamination/Other --

Meets unrestricted release guidelines (MG)

Survey data in file demonstrates residual activity is below the current guideline level for unrestricted release

Exceeds unrestricted release guidelines (EG)

Data indicates some areas or items exceed current guideline level

Not reported in file (NR)

No data are included in Docket File

Detection limit above guidelines (DL)

Survey data are provided, however, the sensitivity of the procedure or the detection limit (MDA) reported is above the current guideline level

Insufficient information (IT)

Data are provided, however, they are incomplete or the measurement technique is not described in sufficient detail to enable evaluation. This code is often a result of data being reported in units which are not directly relatable to guidelines, e.g. cpm or mrad/h for surface activity levels.

Units not compatible with guidelines (UC)

Data are reported in units which cannot be compared with guideline levels: see "insufficient information" above.

2. Survey Documents

No maps of survey locations (MP)

Self-explanatory.

Survey equipment and/or techniques not described (SE)

The information provided is not sufficient to evaluate the adequacy of the survey procedure or to draw conclusions as to the accuracy and adequacy of the data.

Facility areas not surveyed (IS)

Survey data are not provided for certain portions of the facility, considered as potentially affected by licensed activities.

Contaminants not identified (CI)

The principle radionuclides which could remain at the facility are not identified. This information is necessary to establish the guideline levels and evaluate the survey procedures used.

EXPLANATION OF INFORMATION OR STATUS CODES (cont'd)

Specific results not provided (SR)

The File contains general data or summary information from radiological surveys, but does not correlate specific locations where measurements were obtained with associated radiological data.

Guidelines not identified (GI)

The specific guidelines to which the facility was compared when the license was terminated (or transferred) are not identified.

Facility Radiological Status

Meets current guidelines (MG)

Combination of data and other information in File is adequate to conclude that the facility meets the current guideline for unrestricted release.

Low Contamination Potential (LP)

The Docket File does not provide complete documentation supporting a conclusion that the site meets the current guidelines. However, when the information in the Docket File is integrated, the potential for exceeding the current release criteria is low.

License transferred (LT)

The Docket File indicates that all residual radioactive material was transferred to another NRC or Agreement State License or AEC contract.

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	ITT Research Institute
Address:	Chicago, Illinois
Facility:	Homogeneous Reactor
Site Contact:	E. H. Schultz, Director, ITT Research Institute
NRC Contact:	P. A. Morris, AEC, Division of Reactor Licensing
Docket No.:	50-1
License No:	R-3

1. Summary of Operating History

The reactor was licensed on June 12, 1956, to operate at a power level of up to 10 kilowatts. Possession limits were 1300 grams of U-235, as a uranyl sulfate solution, as fuel and 3.5 grams of U-235 in neutron measuring instruments (fission chambers). The Docket File contains no additional design information or operating history. No incidents or spills are described, with exception of a minor fuel spill which occurred during the removal of fuel in preparation for shipment to Savannah River Laboratory. No details on that spill are provided by the licensee; however, an AEC inspection report of September 10, 1971, describes the incident and the actions taken. No residual problem was anticipated as a result of the spill.

2. Request for Termination of the License

The Institute submitted an application for termination of license R-3 on August 9, 1971. The AEC requested additional information and revised application on January 17, 1972. Earlier ITT documents (February 19, 1970, May 15, 1970, September 20, 1970, and December 7, 1970) had described the decommissioning plan in more detail.

3. Dismantling Order Date

The AEC approved the removal of fuel on September 15, 1970. No approval of the dismantling plan was noted in the Docket File.

4. Radiological Status Information

<u>Documentation Identifying Primary Contaminant</u>

In addition to the uranium fuel solution and fission chambers, the licensee's submittal identified fission and activation products as potential residual activity. This included Co-60, Co-59, Fe-55, Cs-137, Ce-144, Pm-147, and Sr-90. Also a plutonium neutron source (size not stated) was identified.

<u>Applicable Guidelines, Supplemental Limits, Exceptions</u>

The applicable guideline at the time of dismantling was not indicated in the Docket File, but it would likely be the "AEC Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use, April 22, 1970."

Licensee's Survey

No information concerning the licensee's survey procedures or specific results is included in the file. There is only a summary statement to the effect that the guidelines have been satisfied.

Measurement, Sampling, and Analyses Procedures

The Docket File does not contain any information on measurement, sampling, and analysis procedures.

Final Survey Data

No specific final survey data are provided in the Docket File.

Confirmatory Survey Results

Inspections of the facility following dismantling and decontamination were conducted on February 14, 1972, and April 10, 1972, by AEC Region III. These inspections noted resolution of discrepancies in fuel possession, identified in a September 10, 1971 inspection. No radiation measurements were performed during these inspections.

In the September 1971 report, it was noted that radiation levels were generally less than 0.1 mR/h with a thin window GM instrument, and levels up to 20-30 mR/h were present at the ends of exposed beam tubes and ports, before the ends were plugged and covered. No follow-up measurements are provided in the Docket File.

Future Use of Facility

The Docket File does not indicate a proposed future use for this facility.

5. License Termination Date

The AEC recommended the reactor license be terminated on April 28, 1972.

Residual activity in components was transferred to by-product material license 12-171-3; residual special nuclear material in fission chambers was transferred to license SNM-49.

6. Federal Register Notice

The notice of termination of the reactor license was published in the Federal Register on May 10, 1972.

7. Hazardous Waste Identified at the Site

No hazardous wastes were indicated in the Docket File.

8. Restrictions/Conditions/Concerns

None noted in Docket File. Residual activity transferred to other licenses.

9. Does the Site Meet Present Criteria?

Insufficient data are provided to evaluate total and removable surface activity relative to criteria. The exposure rate measured by the AEC inspectors indicate that residual activity was present at some shield penetrations; levels were above those currently permitted for unrestricted release. This problem was resolved by transferring residual fission and activation product activity to another facility license.

10. Recommendations

Because residual activity in the facility was transferred to other licenses, it is recommended the uncertainties in the radiological status be addressed as part of the decommissioning for by-product material license 12-171-03 and special nuclear material license SNM-49.

Prepared by: J. D. Berger
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: December 31, 1990

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Naval Research Laboratory
Address:	Washington, D.C. (No additional information provided)
Facility:	Reactor (No additional information provided)
Site Contact:	Earle W. Sapp, Captain - U.S. Navy
NRC Contact:	Peter A. Morris, Division of Reactor Licensing (AEC)
Docket No.:	50-4
License No.:	R-005

1. Summary of Operating History

The research reactor facility of the Naval Research Laboratory was licensed in September 1956 to operate at a power level of up to 100 kw, for purposes of nuclear physics and materials analysis studies. Fuel was enriched uranium; the license allowed possession of 5.0 kilograms of U-235 in fuel element assemblies and 6 grams of U-235 in neutron measuring instruments. The reactor was water moderated. The core was located in a narrow "niche" at one end of the reactor pool. As a result of this configuration, the structural elements in the vicinity of the "niche" contained induced activity, requiring particular attention during the decommissioning activities.

The Docket File contains no information regarding accidents, spills, or fuel failures which could have impacted the radiological status of the facility.

The reactor operation ceased on June 26, 1970, following which the fuel was removed and shipped and decontamination was initiated.

Future Use of Facility

It was expected that the continued use of the building would be in areas of nuclear research.

2. Request for Termination of the License

The Naval Research Laboratory submitted a decommissioning plan to the AEC on June 17, 1970. The plan called for disassembly and shipment of fuel to the Savannah River Plant. Components which were contaminated or contained induced activity were to be disposed of at a licensed waste burial facility. A "niche" which contained induced activity was to be shielded and covered with a 6 inch layer of concrete and posted with a warning sign.

3. Dismantling Order Date

The AEC approved the Dismantling Plan on July 29, 1970.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

According to AEC license R-5, the Naval Research Laboratory was authorized to purchase 5.0 kilograms of enriched uranium (% enrichment not given) in fuel elements and 6 grams of U-235 in measuring equipment. In addition, the activation products Co-60, Zn-65, Cr-51, Fe-59, and Eu-152 were listed in the decommissioning documentation as anticipated contaminants.

Applicable Guidelines, Supplemental Limits, Exceptions

The applicable guideline at the time of dismantling was the "AEC Guidelines of Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use, April 22, 1970."

The decommissioning plan submitted by NRL proposed meeting the NRL Radiological Safety Manual limits which were 1000 dpm/100 cm^2 for removable beta-gamma, 200 dpm/100 cm^2 for removable alpha, and 0.1 mrad/h at 10 cm^2 for fixed beta-gamma.

The total residual activity in the structural material of the "niche" was covered with steel and concrete for shielding and access control and posted.

Licensee's Survey

The Docket File does not contain any information on measurement, sampling, and analyses procedures used by the licensee in conducting the final survey.

The Docket File does not contain any information identifying measurement and sampling locations.

No specific data on results of final radiation measurements by the licensee were provided in the Docket File. The licensee does provide statements in a December 21, 1970 report that the final radiological status was less than 220 dpm/100 cm^2 removable contamination and less than 0.1 mR/h direct radiation.

Confirmatory Survey Results

A confirmatory survey was performed on January 12 and 13, 1971 by AEC Region I inspector, J.P. Stohr. No information concerning survey methods, instruments, specific locations, or specific measurement data is provided in the Docket File. The inspection report states that "remaining activity is within DML guidelines." It was noted that some small spots with direct radiation levels to about 0.4 mR/h at 1 cm distance still remain and the NRL might wish to remove those locations; no followup to this suggestion is documented in the File.

5. License Termination Date

The AEC recommended the reactor license be terminated on April 5, 1971.

Fission chambers containing U-235 were transferred to license SNM-146.

Residual induced activity in the reactor pool "niche" was transferred to byproduct material license 08-01393-02.

6. Federal Register Notice

The notice of intent to terminate the reactor license was published in the Federal Register on March 30, 1971.

7. Hazardous Wastes Identified at the Site

No hazardous wastes were indicated in the Docket File.

8. Restrictions/Conditions/Concerns

No restriction, conditions, or concerns regarding future use of the facility were described in the Docket File.

9. Does the Site Meet Present Criteria?

The removable contamination levels summarized in the licensee's report would meet the present criteria in Table 1 of Regulatory Guide 1.86. Data on total residual surface activity levels are not provided and it is therefore not possible to evaluate conditions relative to those criteria in 1.86. Also, the licensee's exposure rate data of <0.l mR/h are not of sufficient sensitivity to allow comparison with the present criteria for reactor facility decommissioning of 0.005 mR/h (5uR/h) above background.

The Docket File does not adequately identify the extent and location of the surveys performed. No drawings, survey maps, or procedures are included.

10. Recommendations

Because residual activity in the facility was transferred to another Naval Research Laboratory license, it is recommended the deficiencies in reactor license termination actions be addressed as part of the decommissioning for byproduct material license 08-01393-02.

Prepared by: J.D. Berger
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: December 26, 1990

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Battelle Memorial Institute
Address:	Columbus, Ohio
Facility:	Battelle Research Reactor
Site Contact:	J.M. Batch, director
NRC Contact:	D.J. Skovholt, Division of Reactor Licensing
Docket No.:	50-6
License No.:	R-4

1. Summary of Operating History

On August 10, 1956, Battelle was granted a license to operate a research reactor at its Nuclear Science Area site near Columbus, Ohio. The license authorized the possession and use of 5.2 kilograms of contained uranium-235 in fuel element assemblies and 1.63 grams of contained uranium-235 in neutron measuring instruments and a maximum power level of 1000 kilowatts. The reactor was a 2MW modified pool type consisting of a 10 x 10 array of Materials Test Reactor type fuel elements suspended from a movable bridge. On December 31, 1974 the reactor was manually scrammed and the fuel was unloaded in preparation for decommissioning. Three "Unusual Occurrences" were documented during NRC inspections: (1) On December 31, 1974, the reactor was shut down by manual scram and core number 126 was unloaded in preparation for decommissioning of the facility, an incorrect switch was used while transferring cesium-137 contaminated liquid waste from an underground holding tank to an evaporator tank in building JN-1 resulting in a spill of several gallons of the waste. (2) A glass cylinder containing fission gas krypton-85 in the presence of non-radiological gases was broken during an inspection of the container. (3) Soil contaminated with Cobalt-60 (41 pCi/g) and Americium 241 (50 pCi/g) was identified by the licensee in an area six feet downstream from the storm sewer outfall.

2. Request for Termination of the License

A March 6, 1987 application for license amendment by Battelle was considered in the NRC Notice of Proposed Issuance of Order Terminating Facility License as a request for termination.

3. Dismantling Order Date

The NRC issued an Order Authorizing Dismantling on March 19, 1975.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

Primary contaminants are not specifically described. The facility license states that 5.2 kilograms of contained uranium-235 in fuel element assemblies and 1.63 grams of contained uranium-235 in neutron measuring instruments was authorized. NRC investigations noted two receipts of Plutonium-238 as $Pu\text{-}O_2$ containing 22.1 grams and 23.9 grams of total plutonium, a spill of cesium-137 contaminated liquid waste, and a krypton-85 fission gas release.

Applicable Guidelines, Supplemental Limits, Exception

A letter from Battelle to the AEC states their plan to submit a possession only request in accordance with Regulatory Guide 1.86.

Licensee's Survey

Radiological surveys by the Licensee are mentioned in several NRC investigation reports as having disclosed no significant contamination or exposure problems. One report states that quarterly radiation surveys by the Licensee indicated that removable beta activity did not exceed 100 dpm/100 cm^2 and radiation levels were generally less than 2 mR/hr.

Measurement, Sampling and Analyses procedures

The Docket File does not contain any information on measurement, sampling, and analysis procedures.

Final Survey Data

No specific final survey data are provided in the Docket File.

Confirmatory Survey Results

No confirmatory radiological data are provided in the Docket File. An NRC inspection report dated October 21, 1975 indicates that the reactor was dismantled, all spent fuel and radioactive waste was removed from the facility, residual radiation exceeding 5 mR/hr does not exist in any areas that were not physically secured and direct radiation and smear survey results were below Regulatory Guide 1.86 limits. Representative direct radiation surveys of areas which remained posted and secured were viewed; no problems were noted. These areas were identified as: a stall area of the reactor pool, embedded portions of stainless steel beam tubes, and the trench around the reactor pool.

An independent confirmation survey by the Department of Energy was mentioned in a NRC inspection report from October 1982; however, a copy is not included in the Docket File.

An NRC inspection report from September 1989 indicated that the contamination level measured after a Cs-137 spill was less than 1000 dpm/100 cm^2 after clean-up.

Future Use of Facility

The Docket File does not indicate a proposed future use for this facility.

5. License Termination Date

On December 22, 1987, a Termination Order was issued for this license. Residual activity was transferred to License SNM-7.

6. Federal Register Notice

The NRC requested that the Notice of Intent to Issue Order Authorizing Dismantling of Facility be published on February 26, 1975. No Federal Register Notice for License Termination is included in the Docket File.

7. Hazardous Waste Identified at the Site

No hazardous waste was indicated in the Docket File.

8. Restrictions/Conditions/Concerns

No restrictions, conditions, or concerns were noted in the Docket File.

9. Does the Site Meet Present Criteria?

No final or confirmatory survey data are included in the Docket File, therefore, an assessment of the final radiological condition of the site relative to guidelines cannot be made.

10. Recommendations

A radiological survey documenting total and removable surface activity levels, and exposure rates would be required to verify compliance with guidelines. Residual activity was transferred to another license, however, and termination of that license would suffice for this facility.

Prepared By: A.T. Payne
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: February 12, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	North Carolina State College of the University of North Carolina
Address:	Raleigh, North Carolina
Facility:	North Carolina State Homogeneous Reactor
Site Contact:	E.J. Story, Director of Reactor Project
NRC Contact:	E.R. Price, Division of Licensing and Regulation
Docket No.:	50-8
License No.:	R-1

1. Summary of Operating History

The homogeneous reactor at the North Carolina State College facility was licensed on October 1, 1955, to operate up to a power level of 100 watts. The only facility use stated in the file was training related activities. In December of 1962, a letter to the AEC indicated that it was no longer economically feasible to continue operation of the reactor and the space was needed for other projects. The reactor was dismantled and component parts, except shielding blocks, some graphite blocks and the radium-beryllium start-up source, were stored on site until a transfer to Mississippi State University was arranged. The radium-beryllium source was then under the control of a license issued by the State of North Carolina. The license authorized possession of up to 1000 g of contained uranium-235. The majority of this was transferred to the Y-12 facility in Oak Ridge, TN. No operation history was included in the file.

2. Request for Termination of the License

A request for Termination of Facility License No. R-1 was submitted to the AEC on August 23, 1965.

3. Dismantling Order Date

The AEC issued an Order to Authorize Dismantling of the reactor on August 12, 1963.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

No information identifying primary contaminants was provided in the Docket File.

Applicable Guidelines, Supplemental Limits, Exceptions

The Docket File does not identify the applicable guidelines; however, the AEC safety evaluation mentions 10 CFR Part 20 limits for unrestricted areas.

Licensee's Survey

The Docket File does not include any survey data from the Licensee.

Measurement, Sampling and Analyses Procedures

The Docket File does not contain any information on measurement, sampling and analysis procedures.

Final Survey Data

No specific data concerning a final survey by the Licensee is included in the file.

Confirmatory Survey Results

During an inspection by the AEC on August 3-4, 1966, it was noted that: all reactor components were shipped to Mississippi State University; surveys of shipped material showed Ø cpm alpha and 25 cpm beta; surveys made in the area showed background levels only. A safety evaluation by the AEC states that inspectors verified that maximum radiation levels at the site were within 10 CFR Part 20 limits for unrestricted areas and that there were no detectable contamination levels above background on the remaining shield and graphite blocks. No confirmatory radiological data are provided in the Docket File.

Future Use of the Facility

Consideration was being given to the installation of a Pulstar reactor in place of the previous reactor and utilization of the cobalt-60 irradiator pool as the location for an interim training reactor.

5. License Termination Date

A Notice of License Termination was issued by the AEC on September 7, 1966.

6. Federal Register Notice

The Dismantling Authorization Order was sent to the Federal Register for publication on August 12, 1963. The Order for Facility Storage was sent to the Federal Register for publication on December 30, 1964. The Order Authorizing Facility Disposal was published in the Federal Register on December 7, 1965. The Notice of Facility License Termination was published September 16, 1977.

7. Hazardous Waste Identified at the Site

No hazardous waste was identified in the Docket File.

8. Restrictions/Conditions/Concerns

No restrictions, conditions, or concerns were noted in the Docket File.

9. Does the Site Meet Present Criteria?

The Docket File does not provide data concerning the final radiological status of the site, therefore, an assessment relative to guidelines cannot be made.

10. Recommendations

A survey would be required to verify that residual total and removable activity levels and exposure rates satisfy guidelines.

Prepared By: A.T. Payne
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: February 14, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Babcock and Wilcox
Address:	Lynchburg, Virginia
Facility:	Critical Experiment Facility
Site Contact:	A.F. Olsen, Sr. Licensing Administrator
NRC Contact:	C.O. Thomas, Division of Licensing
Docket No.:	50-13
License No.:	CX-10

1. Summary of Operating History

On January 22, 1958, Babcock and Wilcox was licensed to operate a critical facility for experimental purposes. The maximum power level approved by the license was 1000 watts (thermal). Operation continued until September 1983 at which time the fuel was removed and placed in storage on site. By January 29, 1985, all fuel had been shipped from the site. Full length fuel rods were shipped to a DOE facility in Oak Ridge, TN; partial and damaged rod sections and residues from fuel pin cutting operations were sent to National Lead Company, Cincinnati, Ohio. Fuel pins consisted of UO_2 enriched to 2.46% U-235 and clad in stainless steel or aluminum. The file indicates that outside radwaste storage tanks had overflowed at one time and possibly contaminated adjacent soil.

2. Request for Termination of the License

A request was submitted by the Licensee to the NRC on August 7, 1984, to be granted authorization to dismantle and decontaminate the facility to allow for termination of the license. A Dismantling Plan was included with the request.

3. Dismantling Order Date

The NRC issued a dismantling order on April 4, 1985.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The Docket File indicates that fuel pins contained up to 6 kilograms of plutonium in PuO_2-UO_2. The confirmatory survey report by Oak Ridge Associated Universities states that Th-232, U-238, U-235, Co-60 and Cs-137 are the principal gamma emitting radionuclides of interest.

Applicable Guidelines, Supplemental Limits, Exceptions

Regulatory Guide 1.86, Table 1 and exposure rates less than 5 μR/hr at one meter are the applicable guidelines stated in the file.

Licensee's Survey

No information concerning surveys conducted by the licensee prior to the final decommissioning activities is included in the file.

Measurement, Sampling and Analyses Procedures

The Docket File does not contain any information on measurement, sampling, and analysis procedures.

Final Survey Data

The Docket File refers to a final survey report submitted by the licensee to the NRC entitled "License Termination Survey Report for the CX-10 Critical Experiment Facility" and dated June 2, 1986. This report is not included in the Docket File.

Confirmatory Survey Results

The NRC contracted with Oak Ridge Associated Universities to perform a confirmatory radiological survey of the facility. The survey included surface alpha, beta-gamma and gamma scans, measurement of direct and removable contamination levels, and the measurement of radionuclides in soil, gravel, water and paint samples. The report confirmed the findings of the licensee's close-out survey, as well as the compliance with the NRC guidelines established for release for unrestricted use.

Future Use of Facility

The Docket File indicates that Unrestricted Access is desired for this facility.

5. License Termination Date

The NRC approved licensee termination on February 26, 1988.

6. Federal Register Notice

A "Notice of Proposed Issuance of Orders Authorizing Disposition of Component Parts and Terminating Facility License" was published in the Federal Register on September 18, 1984. The file indicates that a copy of the Order Terminating Facility Operating License was sent to the Office of the Federal Register for publication.

7. Hazardous Waste Identified at the Site

No hazardous wastes were identified in the Docket File.

8. Restrictions/Conditions/Concerns

No restrictions, conditions, or concerns were noted in the Docket File

9. Does the Site Meet the Present Criteria?

The information provided by the ORAU confirmatory survey indicates that the site meets the present criteria.

10. Recommendations

The information contained in this report was adequate to evaluate the site. No further action is recommended; however, the file would be more complete if a copy of the "License Termination Survey Report for the CX-10 Critical Experiment Facility" could be located and included.

Prepared by: A.T. Payne
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: February 11, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Babcock & Wilcox
Address:	Lynchburg, Virginia
Facility:	Critical Experimental Facilities
Site Contact:	A. F. Olsen
NRC Contact:	D. J. Skovholt, Division of Reactor Licensing
Docket No.:	50-13 and 50-191
License No:	CX-1 and CX-19

1. Summary of Operating History

Babcock and Wilcox received a license to operate the Split Table Critical Experimental Facility, license CX-1, in March 1957, and the Plutonium Recycle Critical Experiment Facility, License CX-19 in May 1962. Both of these facilities were housed in Bay 1 of Building A at the Lynchburg Research Center. The CX-1 facility operated until October 1961 and the CX-19 facility operated until March 1971.

2. Request for Termination of the License

A request for termination was submitted by Babcock and Wilcox January 25, 1973, and a decommissioning plan was submitted at that time.

3. Dismantling Order Date

A dismantling order was issued by the AEC on March 8, 1973.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

Documentation provided in the AEC inspection report indicated the use of Pu and U fuels. The dismantling log prepared by Babcock and Wilcox revealed the presence of 5 Ci of a Pu-Be source, 1 Ci Po-Be, 65 mCi Co-60 and an undetermined number of Co-60 wires.

Applicable Guidelines, Supplemental Limits, Exceptions

The documentation did not identify the applicable guidelines.

Licensee's Survey

The survey information provided in the Docket File is limited to a summary of exposure notes, and a statement that indicates that no removable activity was detected.

Measurement, Sampling, and Analyses Procedures

Documentation was not provided in the Docket File.

Final Survey Data

Licensee: No data are provided in the Docket File. A statement by BW indicates no detectable fixed or loose contamination identified.

Confirmatory Survey Results

The only confirmatory measurements performed were exposure rate levels in Bay 1. Levels range from .07 to 7 μR/h.

Future Use of Facility

To be used for laboratory facilities.

5. License Termination Date

The AEC issued notification of termination of the license on June 1, 1973. Residual activity was transferred to SNM-778.

6. Federal Register Notice

Notice to dismantle granted on June 3, 1971.

7. Hazardous Waste Identified at the Site

The Docket File did not identify any hazardous waste or hazardous materials. Information provided does not allow for an appropriate assessment.

8. Restrictions/Conditions/Concerns

No anomalies or restrictions were noted in the Docket File.

9. Does the Site Meet Present Criteria?

The documentation did not provide any substantial information to determine if the site meets present criteria.

10. Recommendations

There areinsufficient data to evaluate the radiological status relative to current guidelines. Additional information is required to confirm the accuracy and adequacy of decontamination at the facility. This would include the development of survey documentation exposure rate measurements, and background and baseline documentation. This information may be pursued during actions to terminate license SNM-778.

Prepared by: P. R. Cotten
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: December 20, 1990

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Battelle Memorial Institute
Address:	West Jefferson, OH
Facility:	Critical Assembly-Plastic Reactor Facility
Site Contact:	P.R. Langdon, Asst. Director, Battelle Columbus Laboratories
NRC Contact:	D.J. Skovholt, Division of Reactor Licensing (AEC)
Docket No.:	50-14
License No.:	CX-9

1. Summary of Operating History

On January 16, 1958, Battelle Memorial Institute was licensed to operate a critical facility at their West Jefferson, OH, site. The license and the Docket File documentation do not describe the design of the facility or the radioactive materials used, with exception of a mention that some plutonium sources were used. The approved power level was 200 watts.

No operating history, including spills, incidents, accidents, or fuel failures was provided in the File.

2. Request for Termination of the License

A request for termination of license CX-9 was submitted to the AEC on May 20, 1969. At this time it was noted that the critical assembly facilities had not been operated for several years (since May 1961) and that no source or special nuclear materials were in the possession of Battelle.

No decommissioning plan was submitted with the May 20, 1969 request.

In response to the license termination request (June 12, 1969) the AEC requested Battelle provide additional information on the radiological status of the facility and a plan for disassembly and disposal of components. Battelle provided this information in a response on January 15, 1970.

3. Dismantling Order Date

A dismantling order was not issued by the AEC; the facility had already been dismantled prior to application for license termination.

4. Radiological Status Information

<u>Documentation Identifying Primary Contaminant</u>

No information identifying primary contaminants was provided in the Docket File.

<u>Applicable Guidelines, Supplemental Limits, Exceptions</u>

The Docket File does not identify the applicable guidelines. This action was initiated before the development of the "AEC Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use, April 22, 1970."

Licensee's Survey

The Docket File does not contain any information on measurement and sampling locations.

Measurement, Sampling, and Analyses Procedures

The Docket File does not contain any information on measurement, sampling, and analysis procedures.

Final Survey Data

No specific data are provided in the Docket File. In the January 15, 1970 letter, Battelle notes that all core components and shielding materials have been disassembled and contaminated materials disposed of or are awaiting disposal; plutonium sources were returned to AEC facilities. A radiological survey in March 1969 showed transferable contamination to be less than 10 dpm/100 cm^2 for alpha and less than 40 dpm/100 cm^2 for beta-gamma.

Confirmatory Survey Results

On August 29, 1968, an inspection by F.H. Mutter of AEC Region III noted that the facility had been disassembled and no fuel was stored at the site. No confirmatory radiological data are provided in the Docket File. A Division of Compliance inspection on March 4, 1970 is referred to in the Docket File, but a copy of the inspection is not included.

5. License Termination Date

The AEC issued a letter for termination of the license on May 11, 1970.

6. Federal Register Notice

No Federal Register Notice is included in the Docket File.

7. Hazardous Waste Identified at the Site

No hazardous waste was indicated in the Docket File.

8. Restrictions/Conditions/Concerns

None were noted in the Docket File.

9. Does the Site Meet Present Criteria?

The removable contamination levels reported by the licensee satisfy the present criteria. No data on total activity levels or exposure rates are provided in the Docket File and assessments of these parameters, relative to guidelines cannot, therefore, be made.

The Docket File does not adequately identify the extent and location of the surveys performed. No drawings, survey maps, or procedures are included.

10. Recommendations

Attempts have been made to locate and review the report of the March 4, 1970 AEC inspection of the facility. If that report does not confirm the status of the facility, relative to the present guidelines, an abbreviated survey may be conducted; however, due to the time elapsed since the termination of operations and other uses that have occurred since license termination,it is unlikely that residual contamination from the CX-9 license would still be present. Furthermore, even if any contamination were found, it would be very difficult to associate it with the license.

Prepared by: J.D. Berger
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: December 26, 1990

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Industrial Reactor Laboratories, Inc. N. L. Industries, Inc.
Address:	Plainsboro Township, Middlesex Co., New Jersey
Facility:	Industrial Reactor Laboratory Research Reactor.
Site Contact:	Richard T. Cornfield, Pres. & General Manager
NRC Contact:	Robert W. Reid, Chief, Operating Reactor's #4
Docket No:	50-17
License No:	R-46

1. Summary of Operating History

The research reactor facility was licensed on October 10, 1958, to operate at a power level of up to 100 kw (thermal). The reactor and supporting laboratory facilities were used primarily for basic industrial research and development as well as for providing irradiation services. The pool type reactor was housed in a reinforced concrete parabolic shell. The reactor core was carried from a tower over the pool. The reactor operated a total of 261,809,566 kw hours before experiments and irradiations were terminated in February of 1975. The file describes an inadvertent release of radioactive material to an unrestricted area through a leaking pipe in the waste handling system.

2. Request for Termination of the License

I.R.L. requested the NRC amend the reactor license to "Possession Only" Status on April 21, 1975. An application for termination was submitted on June 12, 1975 and supplemented on November 26, 1976, February 28, 1977, and July 1, 1977.

3. Dismantling Order Date

The dismantling order was approved on September 12, 1975.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

A post decontamination survey report prepared by ATCOR, Inc. states that Co-60 and Cs-137 were the isotopes present.

Applicable Guidelines

A safety evaluation by the NRC indicates that the applicable guidelines were Regulatory Guide 1.86 and 10CFR20

Licensee's Survey

The Dismantling Plan submitted by ATCOR indicates that following final reactor shutdown the reactor core had been unloaded and all reactor fuel and most of the by-product material had been removed from the site. Some radioactive sources were maintained for instrument calibration. The survey to determine radiation levels was performed by ATCOR on May 9, 1975. Radiation dose rates and beta-gamma surface activity levels were measured and ranged from 0.02 to 200 millirem per hour. Loose surface contamination was measured up to 500,000 dpm/100 cm^2.

Measurement, Sampling and Analyses Procedures

The Dismantling plan includes a general description of instrument calibration methods but no specific information concerning measurement, sampling, or analysis procedures. The Post Decontamination report indicates that the following instruments were used:

Eberline gas flow proportional counter Model PAC-4G	Alpha surveys
Eberline Mini Scaler, Model MS-2 with alpha scintillation detector, Model RD-13	Alpha smears
Eberline Model E-120 with end window Model HP-190	Beta-gamma surveys
Hammer Scaler with a GM tube or Eberline Mini Scaler Model MS-2 with Model HP-210 probe	Beta-gamma smears and particulate samples
Eberline Model E-520 with Model HP-177	Gamma surveys

The Post Decontamination report also includes a discussion of calibration techniques and general survey procedures.

Final Survey Data

A final survey report was issued by ATCOR, Inc. on July 1, 1977. The site was surveyed for fixed and removable contamination and gamma radiation levels. Soil and water samples were also analyzed. Descriptions of instrument calibration and conversion of guideline values from dpm/100 cm^2 to mR/h units as applicable to their instrumentation were included. Where radioactivity was determined to be dispersed with a material, a gamma dose rate guideline was developed. Measurement locations were documented on area maps. The average and maximum beta-gamma measurements were 3000 and 15,000 dpm/100 cm^2 respectively. Alpha was < 200 dpm/100cm^2 in all cases. The maximum removable beta activity was 810 dpm/100cm^2. The maximum exposure rate measurement was found to be 0.38 mR/h. Analysis of soil samples indicated maximum values of 9.63×10^{-2} μCi/cc of Cs-137 and $4.18 \times 10^{-}$ μCi/cc of Co-60. Water sample analysis indicated the following maximum levels: 7.2×10^{-7}m μCi/ml for Sr-90, 1.8×10^{-6} μCi/ml for Cs-137, 5.22×10^{-6} μCi/ml for Co-60, and 1.59×10^{-7} μCi/ml for Mn-54. Estimations were done to determine radioactivity remaining in excavations; results were given in total mCi per radionuclide. Leach test analysis was performed on accumulated ground water from tanks determined to contain residual fixed activity above release limits. The results were $\leq 1.5 \times 10^{-7}$ μCi/ml. At the recommendation of the NRC further excavation of contaminated soil in the south corridor was done. The NRC also requested potential biological pathway analyses for four soil use scenarios to determine whether or not a whole body dose of 0.5 rem would be exceeded in one year. The analyses are included in the report.

Confirmatory Survey Results

The Safety Evaluation done by the NRC indicates that a review of the licensee's final survey data was done. The review considered: the guides developed for gamma dose-rate; the sensitivity, ranges and efficiency features of instruments used; survey techniques used to monitor smears, particulate samples, fixed surface contamination;

direct gamma dose-rates. A NRC inspection on September 29-30, 1977, included scans, measurements, and soil and water samples. The survey results confirmed the data presented in the licensee's final survey report.

Future Use of Facility

No information concerning future facility use is included in the Docket File.

5. License Termination Date

The NRC approved license termination on November 4, 1977. Residual byproduct material was transferred to License 29-03686-02.

6. Federal Register Notice

A letter from NRC dated November 4, 1977, states that the termination order was being filed with the Office of the Federal Register for publication.

7. Hazardous Waste Identified at the Site.

No hazardous wastes were identified in the Docket File.

8. Restrictions/Conditions/Concerns

Some residual byproduct material (<20 mCi) will be considered with respect to termination of the Byproduct Material License (29-03686-02).

9. Does the Site Meet the Present Criteria?

The total and removable activity measurements and water sample analyses reported in the licensee's final survey report meet the present guidelines. Some exposure rate measurements were not included and some were taken at distances closer than one meter from the surface and were > 5 μR/hr above background levels. The biological pathway analyses conducted by ATCOR to estimate doses from remaining South Corridor radioactivity are not entirely clear. One scenario is based on soil intrusion occurring no sooner than 50 years from that time. Another scenario deriving a dose rate of 20 mrem/yr; amends their assumption of ingestion rate but does not recalculate a rate based on this amendment.

10. Exposure rate measurements should be taken at one meter from the surface in the facility to determine if values are <5 μR/hr.

Prepared by: A. T. Payne
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: February 20, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Nuclear Development Corporation of America
Address:	Pawling, NY
Facility:	Critical Experiment Facility
Site Contact:	John R. Menke, President, NDC
AEC Contact:	R.L. Kirk, Division of Reactor Licensing (AEC)
Docket No.:	50-23
License No.:	CX-8

1. Summary of Operating History

   The critical experiment facility at the Nuclear Development Corporation of America (NDC) site near Pawling, NY was licensed on January 22, 1958. On May 12, 1961, NDC requested this facility be transferred to the existing Utilization Facility license R-49.

   The Docket File contains no information regarding the design or use of the facility.

2. Request for Termination of the License

   The request for transfer to license R-49 did not require submission of a decommissioning plan.

3. Dismantling Order Date

   Not applicable to transfer only.

4. Radiological Status Information

   Not applicable to transfer only.

5. License Termination Date

   AEC terminated the license effective June 22, 1961.

6. Federal Register Notice

   None noted in the Docket File.

7. Hazardous Wastes Identified at the Site

   Not applicable; transfer only.

8. Restrictions/Conditions/Concerns

   None noted in the Docket File.

9. Does the Site Meet Present Criteria?

   Not pertinent; license transfer only.

10. Recommendations

None

Prepared by: J.D. Berger
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: December 26, 1990

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	General Electric Company
Address:	Pleasanton, California
Facility:	Thermal Critical Assembly: Critical Facility Building
Site Contact:	Mr. Walter H. King, Administrator, Licensing
NRC Contact:	Mr. Donald T. Skovholt, AEC, Director of Reactor Licensing
Docket No.:	50-24
License No:	CX-4

1. Summary of Operating History

General Electric Company (GE) was issued a license August 30, 1957, to possess 8000 lbs. of UO_2 (enriched to 1.6% in U-235), and to operate a Critical Experiment Facility (CEF) at the Vallecitos Nuclear Center. The CEF was located in Building 105. It housed two experimental programs; the Thermal Critical Assembly (TCA) and the Mixed Spectrum Critical Assembly (MSCA). On March 18, 1967, the TCA operated for the final time.

The TCA was used for proof testing fuel bundles for five boiling water reactors, pulsed neutron source experiments, and detailed experiments in a BWR mockup. During the operation of the TCA, no incidents or unusual occurrences were noted. The TCA operated for a total of 1907 hours.

2. Request for Termination of the License

GE submitted an application on May 10, 1967, to possess, but not to operate, the TCA. On January 8, 1968, GE requested an Amendment to CX-4 to authorize further dismantling of the facility components and redefine the physical boundaries of the TCA facility occupied by TCA components. In August 1969 an additional request was submitted. GE received authorization to proceed with dismantling the TCA on September 24, 1969, and submitted an application November 3, 1969, for termination of the license. A report was provided on the completion of the dismantling and disposal of the facility components. All fuel and components from the TCA were transferred to other facilities on site.

3. Dismantling Order Date

The Docket File provides documentation that authorizes GE to possess and store dismantled facility components in the Cell of the CEF dated February 2, 1968, and the AEC granted authorization to dismantle the facility September 24, 1969.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The documentation provided by GE did not identify the primary contaminant.

Applicable Guidelines, Supplemental Limits, Exceptions

GE provided a statement in the documentation that guidelines relative to residual activity have been satisfied, however, no guidelines were identified.

Licensee's Survey

The license did not provide any survey data in the documents in the Docket File. The Inspectors report, dated February 28, 1968, indicates that surveys in the CEF have been conducted on numerous occasions since deactivation of TCA. Elevated alpha activity was detected on the fuel storage vault on January 6, 1968; however, activity levels decayed to below detectable limits in 5 hours.

Measurement, Sampling, and Analyses Procedures

Procedures describing measurements, sampling, and analyses were not provided.

Final Survey Data

The licensee did not provide any final survey data. In an inspection report prepared by R. T. Dodds, AEC reactor inspector, dated February 28, 1968, he indicates that a review of survey records were performed. The records indicated that no detectable radioactivity had been found in the facility.

Confirmatory Survey Results

The AEC Inspector performed a confirmatory survey of the CEF on February 12, 1968. The survey included scans of the CEF Cell, TCA superstructure and components, fuel storage racks, and the vault, using portable beta-gamma and alpha survey meters. Smears for removable activity were collected from these areas. The report states that "no significant radioactivity above background levels was noted as a result of the surveys or the smear swipes."

Future Use of Facility

The space occupied by the TCA was to be used for other purposes as required.

5. License Termination Date

The CX-4 license was terminated by the AEC on December 1, 1969.

6. Federal Register Notice

The termination of license CX-4 was published in the Federal Register December 18, 1969.

7. Hazardous Waste Identified at the Site

Information concerning any hazardous materials or wastes was not provided in the Docket File.

8. Restrictions/Conditions/Concerns

Nothing was noted in the Docket File.

9. Does the Site Meet Present Criteria?

It is difficult to evaluate the radiological status of the site relative to the current criteria due to insufficient data.

10. Recommendations

It was determined from the operating history of the CEF, that two experimental facilities operated in this location under separate AEC licenses. The MSCA was dismantled after the TCA and, the facility occupied by the TCA was to be used for other research activities. The documentation did not identify if other research activities would include the use of radioactive materials. Based on this information it may be difficult to associate residual contamination, if present, to TCA activities. ORAU recommends that a review of the actual survey data and a follow up of the building activities be conducted. If data are not available, a radiological survey should be performed by the licensee, NRC Region V, or an independent contractor.

Prepared by: Phyllis Cotten
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: February 8, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Westinghouse Electric Corp.
Address:	Waltz Mill, PA
Facility:	CRX Facility
Site Contact:	K.R. Schendel, License Administrator, Westinghouse
NRC Contact:	P.A. Morris, AEC Division of Reactor Licensing
Docket No.:	50-34
License No.:	CX-6

1. Summary of Operating History

License CX-6 for the CRX Utilization and Critical Experiments Facility was issued by the AEC on November 25, 1957. The facility was authorized to operate at a maximum power level of 1000 watts (thermal). No additional information regarding the facility design or materials and fuel used is provided in the Docket File. The File also does not contain any information on spills, incidents, fuel failures, etc. which could have impacted the radiological status of the facility.

The decommissioning plan indicates that the building housing this facility and other non-radiological materials will be transferred to the Westinghouse Astronuclear Laboratory, operated under AEC contract NP-1. Special nuclear and byproduct materials will be transferred to another licensee or AEC contractor. Contaminated items and components will be disposed of at a licensed burial facility.

2. Request for Termination of the License

Westinghouse submitted a request for license termination, including a decommissioning plan, on August 4, 1969. No additional information was requested from Westinghouse in support of this action.

3. Dismantling Order Date

The AEC approved the Dismantling Plan on September 30, 1969.

4. Radiological Status Information

The decommissioning plan includes a statement that predecommissioning surveys have indicated no contamination or activation of components exists. Limits presented in the plan are as follows:

25000 dpm/100 cm^2, fixed alpha
1000 dpm/100 cm^2, smearable alpha and beta
0.1 mrad/hr, beta-gamma at 10 cm
(inaccessible surfaces with a potential for contamination will be assumed contaminated.)

No locations of measurements or sampling by the licensee and no specific radiological data are provided in the Docket File. Information describing the licensee's measurement procedures and equipment is not provided. The plan indicates records of survey results will be kept.

The Docket File contains a note regarding an inspection by AEC Region I Compliance Division on November 20, 1969. This note states that the reactor dismantling has been completed per terms of the AEC Dismantling Order. No information regarding confirmatory radiological surveys is included.

5. License Termination Date

The AEC recommended that the license be terminated on December 8, 1969.

6. Federal Register Notice

The notice of license termination was published in the Federal Register on December 17, 1969.

7. Hazardous Waste Identified at the Site

No hazardous waste was indicated in the Docket File.

8. Restrictions/Conditions/Concerns

None noted in the Docket File.

9. Does the Site Meet Present Criteria?

The Docket File contains no data which would permit comparison with present radiological criteria for release. The criteria stated in the licensee's decommissioning plan (see Section 4, above) are higher for total surface activity levels than the present criteria allow; the smearable contamination level of 1000 dpm/100 cm^2 in the decommissioning plan is higher than the current limit for certain radionuclides, such as Sr-90 and Pu-239, which could be considered as potential contaminants (the Docket File does not identify contaminants). No exposure rate data are provided in the Docket File and comparison with the current 5uR/h above background guideline is therefore not possible.

10. Recommendations

There is insufficient data to evaluate the radiological status relative to present criteria; however, the site was operated under AEC contract, following termination of license CX-6. It is unlikely that any residual contamination, if present, could be identified as associated with CX-6 license activities. Documentation on the closeout of the AEC contract activities will serve to confirm the status of this facility.

Prepared by: J.D. Berger
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: December 26, 1990

REVIEW OF TEST AND RESEARCH REACTOR TERMINATED LICENSES

Site:	General Atomic/General Dynamics
Address:	Torrey Pines Mesa, La Jolla, CA
Facility:	CIRCA Facility
Site Contact:	H.B. Fry, Vice President, General Atomic, San Diego, CA
NRC Contact:	R.L. Kirk, Division of Licensing and Regulation, AEC
Docket No.:	50-37
License No.:	CX-3

1. Summary of Operating History

   The CIRCA Facility of General Atomic was licensed on June 26, 1958, to operate for the purpose of conducting tests on the effects of solid poisons on the reactor. The reactor fuel was in metal elements; the core was water moderated in an open tank.

   The license and the Docket File contain no information regarding the type or quantity of fuel authorized by the license.

   No information regarding operating history, including spills, accidents, or fuel failures, is contained in the Docket File.

2. Request for Termination of the License

   A request for termination of the license was submitted to the AEC on January 4, 1960. This request stated that the facility had been dismantled and the special nuclear material placed in storag at the site. It also noted that another facility was under construction and would be operated as an AEC Contractor under Contract AT(04-3)-187.

   No decommissioning plan was submitted.

3. Dismantling Order Date

   None provided in the Docket File.

4. Radiological Status Information

   No information concerning the radiological status of the facility, as determined by the licensee or an agent of the AEC, was provided in the Docket File.

5. License Termination Date

   The AEC issued a termination letter to General Atomic on March 15, 1960.

6. Federal Register Notice

   The notice of license termination was published in the Federal Register on March 19, 1960.

7. Hazardous Wastes Identified at the Site

   No hazardous waste was indicated in the Docket File.

8. Restrictions/Conditions/Concerns

None indicated in the Docket File.

9. Does the Site Meet Present Criteria?

The Docket File contains no information on which to base an assessment of the radiological status, relative to criteria.

10. Recommendations

The Docket File does not provide sufficient information to evaluate the radiological status. Additional information is necessary to verify the adequacy of decommissioning and decontamination actions at the facility.

If the facility was later operated under an AEC contract, the decommissioning documentation for that activity may provide sufficient information on which to base a decision as to ultimate disposition.

Prepared by: J.D. Berger
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: December 26, 1990

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site: U.S. Naval Postgraduate School
Address: Monterey, CA
Facility: AGN 201 Reactor
Site Contact: R.W. McNitt, U.S. Navy, Monterey, CA
NRC Contact: D.J. Skovholt, AEC Division of Reactor Licensing
Docket No.: 50-43
License No.: R-11

1. Summary of Operating History

The license issued on April 29, 1957, authorized possession of 666 grams of U--235 as fuel and a 10 mCi RaBe neutron source. No information on the design or operating history of the reactor is provided in the Docket File. The File contains no information on spills, incidents, or fuel failures.

2. Request for Termination of the License

A request to dismantle and store the reactor was submitted on March 16, 1971; a request for license termination, including a plan for dismantling and disposition of the reactor to California State Polytechnic College was submitted on August 25, 1971.

3. Dismantling Order Date

The AEC approved the Dismantling Plan on November 17, 1971.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

Primary contaminants are not described; however, based on the facility use, uranium and mixed fission and activation products would be the expected potential contaminants.

Applicable Guidelines, Supplemental Limits, Exception

The applicable guideline at the time of dismantling was not indicated in the Docket File, but it would likely be the "AEC Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use, April 22, 1970."

Licensee's Survey

The licensee states that a survey was conducted using an AN/PDR-27 end window GM instrument, having a low-scale range to 0.5 mR/h. Wipe tests were also conducted and counted in a Picker NaI well counter.

The August 2, 1972 survey of the facility gamma levels are less than 0.025 mR/h and removable contamination is a maximum of 139 cpm (background 118.4 cpm). A statement is provided that removable activity is well below the limits specified for releasing the building for public use.

Individual locations of measurements, procedures, or specific data are not provided in the File.

Confirmatory Survey

AEC Region V conducted on inspection on August 28, 1972. The report notes that licensee survey results are documented in the reactor logbook, transferred to California State Polytechnic College. The AEC assessment is that no radiation or removable contamination above background levels remains. No independent confirmatory measurements by the AEC are indicated in the File.

5. License Termination Date

The AEC recommended that the license be terminated on October 12, 1972.

6. Federal Register Notice

The Docket File did not include a copy of the Federal Register Notice of license termination.

7. Hazardous Wastes Identified at the Site

No hazardous wastes were indicated in the Docket File.

8. Restrictions/Conditions/Concerns

None noted in the Docket File.

9. Does the Site Meet Present Criteria?

Insufficient radiological data are available to demonstrate that exposure rates and total surface activity levels satisfy present criteria. The removable contamination levels given by the licensee do meet the current guidelines.

10. Recommendations

A survey would be required to verify that building exposure rates are less than 5 uR/h above background and that residual total surface activity satisfies guidelines.

However, it is considered unlikely that such a survey, conducted almost 20 years after the reactor operation and in a building likely used for other activities, would be very meaningful.

Prepared by: J.D. Berger
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: December 26, 1990

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Atomics International/North American Aviation, Inc.
Address:	1600 VanOwen Street Canoga Park, California
Facility:	Model L-47 Research Reactor
Site Contact:	T.F. Humphrey, Director Contract Administration
NRC Contact:	Lyall Johnson, Division of Licensing and Regulation
Docket No.:	50-50
License No.:	R-19

1. Summary of Operating History

On August 5, 1957 Atomics International, a division of North American Aviation, Inc., received a license to operate a research reactor at their Canoga Park, California site. The license authorized possession and use of 1200 grams of contained uranium-235 in the form of uranyl sulfate as fuel for operation. The approved power level was 5 watts. The Docket File contains no information concerning reactor design or operating history. The reactor was dismantled by May 26, 1958, and placed in storage.

2. Request for Termination of the License

A request for termination of license R-19 was submitted to the AEC on May 26, 1958.

3. Dismantling Order Date

The Docket File contains no documentation of a Dismantling Order.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

No information identifying primary contaminants is provided in the Docket File.

Applicable Guidelines, Supplemental Limits, Exceptions

The Docket File does not identify the applicable guidelines.

Licensee's Survey

The Docket File contains no information concerning a facility survey.

Measurement, Sampling and Analysis Procedures

The Docket File contains no information on measurement, sampling, and analysis procedures.

Final Survey Data

The Docket File contains no information concerning a facility survey.

Confirmatory Survey Results

The Docket File contains no information concerning a confirmatory survey.

Future Use of Facility

The Docket File does not state any future intentions for facility use.

5. License Termination Date

The AEC issued a License Termination Notice on June 30, 1958.

6. Federal Register Notice

Federal Register Notice of Termination of Utilization Facility License was published on July 9, 1958.

7. Hazardous Waste Identified at the Site

No hazardous waste was indicated in the Docket File.

8. Restrictions/Conditions/Concerns

No restrictions, conditions, or concerns were noted in the Docket File.

9. Does the site Meet the Present Criteria?

No data is provided in the Docket File to indicate the final radiological condition of this facility, therefore, an assessment of compliance to guidelines cannot be made.

10. Recommendations

A survey would be required to verify that residual total and removable activity levels and exposure rates satisfy guidelines.

Prepared: A.T. Payne
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: February 13, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Oklahoma State University
Address:	Stillwater, Oklahoma
Facility:	AGN Nuclear Reactor
Site Contact:	R. B. Kamm, President, Oklahoma State University
NRC Contact:	D. J. Skovholt, Division of Reactor Licensing
Docket No.:	50-58
License No:	R-22

1. Summary of Operating History

Oklahoma State University acquired a license to operate a 100 Mwatt AGN201 nuclear reactor August 26, 1957. The license authorized the University to receive 700 grams of uranium 235 for the purpose of operating the reactor. The reactor facility was originally located in Engineering Unit No. 2. The reactor operated in this area until May 27, 1965, and on June 11, 1965, the reactor was relocated to the Engineering North Building. This area consisted of rooms B06A, B06B, and B06C. The final report provided by OSU indicates that the reactor was never operated in this location since the control rods were not installed in the reactor after relocating.

2. Request for Termination of the License

OSU submitted an application for termination of the license R-22 on May 26, 1971. OSU submitted a dismantling plan along with the application.

3. Dismantling Order Date

The AEC authorized the dismantling of the facility and the transfer of the Reactor to Tuskegee Institute in Tuskegee, Alabama on July 13, 1972.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The final survey report prepared by OSU, January 23, 1974, did not identify the primary contaminant.

Applicable Guidelines, Supplemental Limits, Exceptions

The inspectors report dated October 16, 1972, specified that residual activity levels were compared with the Commissions "Guidelines for Decontamination of Facilities and Equipment, etc." issued April 22, 1970.

Licensee's Survey

During the dismantling activities, radiation levels were measured and recorded. The surfaces of all of the reactor components were smeared and the smears analyzed for gross beta-gamma and alpha removable contamination. Surveys of the building surfaces, i.e., floors and walls, are not included in information provided in the Docket File.

Measurement, Sampling, and Analyses Procedures

In the final survey data submitted in the document dated January 23, 1974, OSU identified the use of an end window GM (window thickness 1.9 mg/m^2) for direct measurements and using a tracer lab monitor. Information of the sampling and analyses was not provided. The floor was decontaminated by using a scrubber and floor tiles were removed.

Final Survey Data

The final survey report indicates that the maximum removable activity level from the floor was 6 dpm alpha/100 cm^2 and 9 dpm beta/100 cm^2.

Confirmatory Survey Results

An independent survey was conducted by the NRC inspector during an October 16, 1972, inspection. This included smears of the surfaces of reactor components and fuel plate. Water samples from the shield tank and thermal column water were also collected from the licensee. The results of the smears indicate levels below an established background with the exception of the fuel plate. The removable activity levels were reported as 267 dpm/alpha/fuel plate and 76 dpm beta-gamma fuel plate. Concentrations in the water did not differ from those levels reported by the licensee.

Future Use of Facility

The Engineering Unit No. 2 contains offices and classrooms. The Engineering North Building also consists of offices and classrooms.

5. License Termination Date

The AEC terminated the license March 21, 1974. The reactor components and fuel were shipped to Tuskegee Institute March 23, 1973.

6. Federal Register Notice

An order for publication of the notice of termination of license R-22 was issued March 21, 1974. The Docket File does not contain a date the notice was published.

7. Hazardous Waste Identified at the Site

No hazardous wastes or material were identified.

8. Restrictions/Conditions/Concerns

None were identified.

9. Does the Site Meet Present Criteria?

The evidence provided by the licensee of residual radioactivity levels of the floor in Room B06A would meet the present criteria. However, an accurate assessment of the facilities involved in the reactor activities is difficult to discern from the information provided.

10. Recommendations

Information is not available for the area in which the reactor was originally operated. A survey of this area in addition to Room B06A would be required to confirm that residual total surface activity and building exposure rates satisfy the guidelines. This activity could be performed by the University, federal agency, or its designee.

Prepared by: P. R. Cotten
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: January 31, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site: U. S. Naval Hospital
Address: Bethesda, Maryland
Facility: AGN 201 Nuclear Reactor
Site Contact: G. M. Davis, Captain, U. S. Navy Medical Corps
NRC Contact: R. S. Byrd, Division of Reactor Licensing
Docket No.: 50-60
License No: R-27

1. Summary of Operating History

In March 1957, the AEC granted the U.S. Naval Hospital in Bethesda, MD a license to possess and operate the AGN 201M model nuclear reactor. The license entitled the Naval Hospital to possess and use up to 700g of U-235. No operating history was provided in the Docket File. Operation of the reactor was discontinued in April 1962.

2. Request for Termination of the License

A request for termination of the license was submitted August 30, 1963. Final disposition of the reactor would be at the New York University located in New York City.

3. Dismantling Order Date

The Docket File did not contain a dismantling order issued by the AEC.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The primary contaminant was not identified in any of the documentation provided in the Docket File. However, uranium, mixed fission and activation products would be expected potential contaminants.

Applicable Guidelines, Supplemental Limits, Exceptions

Applicable guidelines for release of an area to unrestricted use was not identified. The actions performed by the U.S. Naval Hospital preceded the AEC Guidelines of April 22, 1970.

Licensee's Survey

The documentation provided by the Naval Hospital does not indicate any incidents of spills. Survey data provided by the licensee indicate that no detectable beta or gamma activity was identified in excess of background (0.01 μR/h). Smears were obtained from the reactor room and counted with a Nuclear Chicago scaler.

Measurement, Sampling, and Analyses Procedures

No procedures were described for measurement, sampling, or analyses.

Final Survey Data

All survey results were reported in terms of μCi/100 cm^2. Smears were collected from the reactor room. The licensee provided a map of smear locations.

Confirmatory Survey Results

The AEC inspector performed a visual inspection of the former reactor room and made copies of the survey data.

Future Use of the Facility

Future use of the reactor room is unknown.

5. License Termination Date

The AEC terminated the license on June 29, 1965, and provided title of the reactor and components to the New York University.

6. Federal Register Notice

The termination notice was published July 1, 1965.

7. Hazardous Waste Identified at the Site

No hazardous wastes were identified at [illegible] The operations and information provided suggest that this was not a si[illegible]t factor.

8. Restrictions/Conditions/Concerns

No anomalies or conditions were noted.

9. Does the Site Meet Present Criteria?

The licensee indicated that a survey was performed at the time the reactor was moved, however, the data has been misplaced. Survey data that are available, indicate the location of smear samples and the results of the analyses. The activity reported is in units of μCi/100 cm^2 The background exposure rate level determined by the licensee was 0.01 μR/h. This would meet the present criteria.

10. Recommendations

A survey of the reactor room for total surface activity would be required to determine if the area would meet current guidelines.

Prepared by: P.R. Cotten
Oak Ridge Associated Universities
Oak Ridge, TN 37830
Date: December 20, 1990

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site: University of Akron
Address: Akron, Ohio
Facility: AGN 201 Nuclear Reactor
Site Contact: W. M. Petry, University of Akron
NRC Contact: D. S. Skovhott, AEC Division of Reactor Licensing
Docket No.: 50-64
License No: R-24

1. Summary of Operating History

The University of Akron (UOA) was licensed in September 1957 to operate and to possess 700 grams of contained U-235 for use in the AGN 201 nuclear reactor. UOA operated the reactor until 1966. The maximum power level utilized was approximately 45 Mwatts. The reactor had a total power output of 100 Mwatts. Historical information on the operating history is available only for the years prior to final shut down of the reactor.

2. Request for Termination of the License

The University submitted an application for termination February 9, 1967. On January 18, 1967, the University of Akron submitted a plan to dismantle and transfer the reactor to Georgia Institute of Technology (Docket No. 50-276).

3. Dismantling Order Date

The University of Akron and Georgia Institute of Technology were issued authorization to dismantle the reactor July 28, 1967. The University applied for permission to dismantle February 9, 1967. The University advised the AEC that the UOA had to hold up the transfer decision until Georgia Tech committed to receiving the reactor.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The primary contaminant was not identified in information provided in the Docket File.

Applicable Guidelines, Supplemental Limits, Exceptions

Applicable guidelines were not identified in the information provided in the Docket File.

Licensee's Survey

The licensee did not perform the survey of equipment and the facility where the reactor was located. Swipe tests were performed by Mr. Hoyt, Health Physicist from GIT under the supervision of Mr. Petry, Reactor Supervisor, UOA. The information provided indicates that a survey was performed but no data were available for review.

Measurement, Sampling, and Analyses Procedures

No documentation was provided in the Docket File.

Final Survey Data

The licensee did not perform any surveys during the decontamination activities. These duties were accepted by Georgia Tech. No data were provided.

Confirmatory Survey Results

The inspector (G. Fiorelli) was unable to conduct an independent survey of the area where the reactor was located since a new 4-inch concrete floor had been poured subsequent to shipping the reactor. The building was being partitioned and painted at the time of the inspection.

The inspector did review the survey data developed by the Georgia Tech health physicist. He indicates that no contamination measurements of the reactor components or the area where the reactor was located was above normal background. No information was provided as to normal background levels.

Future Use of Facility

The future use of the facility appears to have been for classroom use. However, this is not stated in the Docket File.

5. License Termination Date

The AEC terminated license R-24 on October 9, 1967.

6. Federal Register Notice

A notice of termination was published in the Federal Register, October 19, 1967.

7. Hazardous Waste Identified at the Site

The Docket File did not contain information that discussed or identified any hazardous wastes. Due to the type of reactor facility it is unlikely that hazardous waste would be associated with this type of operation.

8. Restrictions/Conditions/Concerns

The shield tank water was drained to the sewer in July, 1966; no measurement of the radioactivity of the waste was determined. However, the licensee felt assured that the activity in the water was not above background levels.

9. Does the Site Meet Present Criteria?

The information available did not provide adequate documentation to make an appropriate judgement of the radiological status of the facility.

10. Recommendations

A review of the survey information, if obtainable, would be adequate to make an appropriate judgement as to the radiological status of the facility. Due to the modifications made to the facility, a survey would probably not provide meaningful information.

Prepared by: P.R. Cotten
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: January 31, 1991

REVIEW OF TEST AND RESEARCH REACTOR TERMINATED LICENSES

Site: National Aeronautics and Space Administration (NASA)
Address: Cleveland, Ohio
Facility: Zero Power Reactor Facility
Site Contact: B. T. Lundin, Director, NASA
NRC Contact: D. S. Skovhott, AEC Division of Reactor Licensing
Docket No.: 50-75
License No: CX-13

1. Summary of Operating History

NASA was granted a license January 5, 1959, to operate a homogeneous Zero Power Solution type research reactor. The reactor facility was located at NASA's Lewis Research Center in Cleveland, Ohio. The reactor was operated in the Zero Power Reactor Facility located in the basement of the Materials and Stress Laboratory Building. This area consisted of 4 rooms; a control room, a solution room, a reactor room, and a locker room. The license allowed NASA to possess up to 3000 grams of Uranium 235 in the form of uranyl fluoride. The licensed maximum steady state thermal power level was 100 watts.

Zero Power Reactors (ZPR) I, operated until February 23, 1973. The reactor operated for only 6 days during 1973. All operations were pulsed neutron source experiments. The total integrated power output during the operational life of the reactor was over 5700 watt-hours.

2. Request for Termination of the License

NASA submitted a request for termination February 1, 1973. The request included a plan for dismantling the reactor facility.

3. Dismantling Order Date

The AEC issued to NASA an order to dismantle the Zero Power Reactor Facility on March 30, 1973.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

NASA performed contamination surveys of the facility during the dismantling process, however, the primary contaminant was not identified.

Applicable Guidelines, Supplemental Limits, Exceptions

Residual activity levels at the ZPR were compared with the AEC Guidelines for facilities and equipment issued April 22, 1970.

Licensee's Survey

The licensee performed surveys and continuous monitoring during the dismantling period in 27 locations in the reactor room and 11 locations in the solution room. Surveys were performed weekly during dismantling operations until all fuel had been removed. The control room, hallway and adjacent annex were surveyed weekly in 13 locations. No smears collected in these areas exceeded the limits of 10 dpm/100 cm^2 alpha or 400 dpm/100 cm^2 beta-gamma during this survey period.

Measurement, Sampling, and Analyses Procedures

A brief description of the decontamination procedures and survey procedures were provided in the Final Radiation Survey Report prepared by NASA.

All items were removed from the ZPR facility prior to decontamination. All surfaces and remaining structures in the reactor room were sandblasted and the steel floors wet mopped. Surfaces were scrubbed in the solution room.

Smears were collected and counted using a thin window (<150 $\mu g/cm^2$) proportional counter. Total contamination measurements were obtained using a portable alpha scintillation counter (window thickness 1.5 $\mu g/cm^2$). The alpha scintillation detectors were calibrated using a Pu-239 source.

Final Survey Data

The licensee performed a final survey of the facility. The information provided includes figures of the floors and walls of each room sample and measurement location, and survey data. Actual data were only provided for the solution room hold-up tank, piping lights and duct on this area, and for several miscellaneous items in the reactor room.

Total activity levels range from 0 to 5600 dpm/100 cm^2 and removable activity ranged from 0 to 400 dpm/100 cm^2. A lower limit of detection was not provided in the data.

Confirmatory Survey Results

The Docket File did not indicate that a confirmatory survey had been performed.

Future Use of Facility

The future use of the facility was not developed in the contents of the Docket File.

5. License Termination Date

The AEC terminated license CX-13 November 13, 1973.

6. Federal Register Notice

Notice of publication in the Federal Register was not provided in the Docket File.

7. Hazardous Waste Identified at the Site

Hazardous Wastes or use of hazardous materials were not discussed in the Docket File. It is difficult to discern if such a problem existed.

8. Restrictions/Conditions/Concerns

None

9. Does the Site Meet Present Criteria?

Residual activity levels reported in the final survey would be in accordance with present release criteria for residual uranium activity. However, since data were not reported in the tables provided for all the areas surveyed, it is difficult to confirm that the control room, reactor room, solution room, locker room and any access hallways meet the present criteria.

10. Recommendations

It may be difficult to obtain the original survey data for review. A survey of the former ZPR facility by the current occupant, NRC Region III, or agent of the NRC, would determine whether the radiological status satisfies the current guidelines.

Prepared by: P. R. Cotten
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: January 29, 1991

REVIEW OF TEST AND RESEARCH REACTOR TERMINATED LICENSES

Site:	University of California
Address:	Berkeley, California
Facility:	AGN 201 Nuclear Reactor
Site Contact:	H. Mark
NRC Contact:	R. T. Dodds, AEC Division of Reactor Licensing
Docket No.:	50-84
License No:	R-30

1. Summary of Operating History

In 1957, the University of California at Berkeley (UCB) obtained a license to operate a nuclear reactor Model AGN-201 manufactured by Aeropt-General Nucleonics (AGN). The actual dates of operation are unclear from the information provided in the Docket File. This license allowed UCB to possess and use up to 700 g of U-235. The AGN reactor was located in Room 165 in Cory Hall on the UCB campus.

2. Request for Termination of the License

UCB submitted a request to the AEC for termination of the license R-30 on July 25, 1966. The decommissioning plan provided to the AEC indicates that the reactor would be transferred to the University of New Mexico. The plan does not provide sampling plan analyses procedures.

3. Dismantling Order Date

AEC authorized the dismantling and transfer of the AGN Reactor on June 17, 1966.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The primary contaminant was not identified in the information provided in the Docket File.

Applicable Guidelines, Supplemental Limits, Exceptions

The Docket Files does not identify any applicable guidelines.

Licensee's Survey

Minimal survey data were provided in the Docket File. The reactor fuel disc was surveyed. The most active reading observed was 500 rem/hr on the fuel disc and 60 rem/hr on the bird cage.

The AEC inspector at the facility requested that all components be surveyed and swiped. The lead shield rings and plugs were noted to have levels of 0.1 to 0.4 rem/hr at contact.

A survey was conducted in the reactor room. The AEC reviewed the survey records and indicated that no readings were in excess of 25 cpm. The reactor room was swept and then mopped.

Measurement, Sampling, and Analyses Procedures

Information provided in the AEC inspectors' report indicates that scanning and measurements on components were obtained using a GM detector. No indication was given as to how dose rates were obtained or what type of radiation was measured to obtain the value of 25 cpm.

Final Survey Data

Sufficient final survey data of the reactor room is not available in the Docket File. UCB stated that no detectable activity was measured.

Confirmatory Survey Results

No confirmatory measurements were obtained by the AEC inspector. Inspection reports discuss survey data.

Future Use of Facility

Future use of the facility was not identified in the Docket File.

5. License Termination Date

Termination of the license was authorized August 23, 1966.

6. Federal Register Notice

The Federal Register Notice of the termination of license R-30 is dated August 31, 1966.

7. Hazardous Waste Identified at the Site

No hazardous wastes were identified in the Docket File. The nature of the operations does not suggest this would be a significant factor.

8. Restrictions/Conditions/Concerns

UCB was cited for one violation to the dismantling order. Water from the reactor shield water tank was dumped to the sanitary drain prior to being sampled. UCB indicates in their reply to the AEC that nothing was detectable in the samples collected from the sanitary drain after the discharge.

9. Does the Site Meet Present Criteria?

The Docket File does not contain sufficient data for the reactor room which would permit comparison with present release guidelines.

10. Recommendations

A review of the survey records if maintained by UCB may be sufficient to determine if survey results are adequate to determine if current guidelines can be met. If this information is not available, a survey would be required to verify that building exposure rates are less than 5 μR/h above background and that residual surface activity criteria is satisfied. However, it is most likely that a survey of a facility of this nature, that operated 20 years ago would provide any meaningful information.

Prepared by: P. R. Cotten
Oak Ridge Associated Universities
Oak Ridge, TN 37830
Date: December 20, 1990

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site: Westinghouse Electric Corporation
Address: Waltz Mill, Pennsylvania
Facility: Critical Experiment Station Facility
Site Contact: Karl R. Schendel, Westinghouse License Administrator
NRC Contact: Donald J. Skovholt, Division of Reactor Licensing
Docket No: 50-87
License No: CX-1

1. Summary of Operating History

The Critical Experiment Station (CES) Facility was licensed June 17, 1958, to operate at power levels up to 100 watts for performing criticality experiments. License CX-11 authorized the possession and use of 21 Kg of contained U-235 as fuel. On September 24, 1970, Westinghouse requested authorization to dismantle and relocate the facility from Waltz Mill to Zion, Illinois. The order authorizing dismantling was issued February 12, 1971. An October 21, 1971, inspection confirmed that the facility had been moved to Zion. On January 19, 1972, Westinghouse reported that all special nuclear material had been transferred. License CX-11 was terminated on January 26, 1972.

No unplanned release of radioactive material or disposal under 10 CFR 20.304 were noted in the docket file.

Future Use of the Facility

The CES facility building was transferred to the Westinghouse Advanced Reactor Facility under SNM-770.

2. Request For Termination of License

The December 18, 1970 supplement to the September 24, 1970, request to dismantle the facility was referenced in the termination order. However no specific request to terminate the license was submitted.

3. Dismantling Order Date

The Dismantling Order was issued February 12, 1971.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

No information on the primary contaminant at the facility was provided other than the material authorized in the operating license.

Applicable Guidelines, Supplemental Limits, Exceptions

The October 12, 1971, "closeout" inspection report referenced the "Guidelines For Decontamination Of Facilities And Equipment Prior To Release or Unrestricted Use Or Termination Of Licenses For Byproduct, Source, Or Special Nuclear Material April 2, 1970."

Licensee Survey

A January 19, 1972, letter from Westinghouse states that the structures at the Waltz Mill site were surveyed (airborne, radiation, and smear survey) and the only item above background was the core tank. There is no documentation on the measurement, sampling and analysis procedures used by the licensee. The NRC closeout inspection mentions surveys that were reviewed with no reference to a particular document.

Confirmatory Survey Results

An October 21, 1971, closeout inspection by E.R. Loibl, Region I, verified the transfer of the facility to Zion. The fuel (40 elements), 2 Pu-Be sources and 2 fission chambers were transferred to the hot cell room at the Westinghouse Advanced Reactor Division facility at Waltz mill pending issuance of the Zion facility license. Accountability for the above material along with the CX-11 building, control rod drive support platform, reactor tank, reactor dump tank, associated piping, and any residual radioactivity remaining were transferred to special nuclear materials license SNM-770.

The inspection report states that smear survey records indicated "spreadable contamination" less than "50 dpm". Independent smear surveys performed by the inspector and analyzed by HASL agreed with licensee results. There is no documentation on the measurement, sampling, or analysis procedures used by the inspector.

5. License Termination Date

The License Termination Order was issued January 26, 1972.

6. Federal Register Notice

The termination order was published in the Federal Register on February 4, 1972.

7. Hazardous waste Identified at the Site

No hazardous wastes were identified in the docket file.

8. Restrictions/Conditions/Concerns

No restriction, condition, or concerns regarding future use of the facility were described in the docket file.

9. Does the Site Meet the Present Criteria?

The 50 dpm reported in the inspection report is the only survey result in the docket file. There is no reference to the area over which the sample was collected or whether it refers to alpha, beta-gamma, or both. There is no data on the level of fixed contamination remaining or exposure rates. The data presented does not supply enough information to determine if the acceptable surface contamination levels of Regulatory Guide 1.86 or the exposure rate limit of 5 μR/hr were met.

10. Recommendations

The residual radioactivity remaining at the CX-11 facility was transferred to special material license SSN-770. This radioactive material should be addressed during the decommissioning of that license.

Prepared by: David N. Fauver
Division of Low-Level Waste Management and Decommissioning
Office of Nuclear Material and Safeguards

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Westinghouse Electric Corporation
Address:	Zion, Illinois
Facility:	Zion Nuclear Training Reactor
Site Contact:	A.J. Nardi, Manager
NRC Contact:	T.S. Michaels, Office of Nuclear Reactor Regulation
Docket No:	50-87
License No:	R-119

1. Summary of Operating History

The critical Experimental Station Reactor located in Waltz Mill, Pennsylvania, received license CX-11 to operate on June 17, 1958 at a maximum power level of 100 watts. The facility was used to conduct low power physics tests, functional component tests and experimental apparatus environmental studies. In 1964, partial redesign was carried out to achieve more flexible operations and obtain a higher maximum power level (10kw). The reactor was moved to Zion, Illinois in 1971 and relicensed as R-119, Nuclear Training Reactor (NTR), on January 28, 1972. The NTR was described as a light-water moderated, graphite reflected, pool type reactor. The NTR core consisted of 19 fuel elements of high enrichment uranium. The core was arranged in a hexagonal configuration, surrounded by graphite reflectors; light water served as the coolant and supplemented the graphite as a moderator. The authorized maximum power level was 10kw (thermal). Reactor utilization was 80% training of commercial power plant operators in fundamental theoretical principles of reactor operation, 15% maintenance and 5% irradiation experiments. The reactor history indicates that the system was very dependable, with no single major component failures occurring, no detectable releases of radiation to the environment, no radiation spills and no areas of high contamination resulting from operation. There were two incidents of dropped fuel elements, neither of which resulted in facility contamination or damage to the fuel element. The reactor has been shut down since February of 1987.

2. Request for Termination of the License

A request for termination of license R-119 was submitted to the NRC on July 8, 1987.

3. Dismantling Order Date

The NRC issued an Order Authorizing Dismantling of Facility and Disposition of Component Parts on January 29, 1988.

4. Radiological Status Information

<u>Documentation Identifying Contaminants</u>

The confirmatory survey report submitted by Oak Ridge Associated Universities identifies the primary radionuclides of concern as: uranium, long lived fission products (primary Cs - 137), and long lived activation products (primarily Co - 60).

Applicable Guidelines, Supplemental Limits, Exceptions

Guidelines for residual total and removable activity were determined to be 15,000 dpm/100 cm^2 area and 5,000 dpm/100 cm^2 average over 1 m^2 for total activity and 1000 dpm/100 cm^2 for removable activity. Exposure rate measurements were required to be <5 μR/h above background.

Licensee's Survey

The reactor core status survey performed prior to dismantling indicated that elevated spots were located near steel bolts holding the reactor segments together. The most elevated location inside the reactor tank measured 82.9 μR/hr. Air samples showed maximum activities of 6.52 E-13 μCi/ml alpha and 1.04 E-11 μCi/ml beta-gamma. Water samples from the reactor tank indicated maximum activities of 0.125 dpm/ml alpha and 19.6 dpm/1000 ml beta-gamma. All readings taken in the dump tank were <20 μR/hr. A contact reading on the demineralizer indicated 15.6 μR/hr. Smears from the demineralizer and it's pre-filter and pump, the moderator fill pump, dump tank and valve indicated activities less than the lower limit of detectability. Area surveys in the machine shop were approximately 19.3 μR/hr. A contact reading of the air-compressor receiver tank was <20 μR/hr. A contact reading on the sump collection showed <15 μR/hr. A survey of areas isolated from the reactor room: control room, Health Physics lab, classroom, and office area indicated a maximum exposure rate of 14.2 μR/hr and smear results were all less than LLD for alpha and beta activities.

Measurement, Sampling and Analyses Procedures

General methods for measurement, sampling, and analysis procedures are outlined in the Licensee's Decommissioning Final Report and in the confirmatory report submitted to the NRC by ORAU.

Final Survey Data

A letter from Westinghouse to the AEC dated January 19, 1972, indicated that the CES facility had been dismantled and all special nuclear materials were transported to the Zion, Illinois site as authorized by CPRR-113. A survey including airborne radiation and smear determinations was conducted. The only item found to have greater than background activity was the "old core tank". The tank was removed for use on a portion of the Waltz Mill site where it would be covered along with the former facility structures by materials License SNM-770.

The Licensee performed a detailed survey following dismantling of the Zion facility. Surfaces were surveyed for total alpha and beta-gamma activity, removable activity, and exposure rate measurements. Surveys of miscellaneous objects such as pumps, pipes, valves, drain lines, and vent systems were also performed. The survey was separated into areas I-III based on the probability of contamination for the area, with Section I consisting of the lowest probability areas and Section III including areas where radioactive materials were normally utilized and stored. The maximum exposure rate measured was 18.6 μR/hr. Maximum fixed activity was found to be 139 dpm/100 cm^2 alpha and 7883 dpm/100 cm^2 beta-gamma. Maximum removable activity was determined to be 5.6 dpm/100 cm^2 alpha and 145.4 dpm/100 cm^2 beta. A list of survey instrumentation is provided along with a statement that standardized facility sources were used for efficiency calibration.

The following instruments were used for this survey:

Alpha:	Eberline PAC-15A, AC-8 Technical Associates Staplex
Beta-Gamma:	Eberline E-530, RM-15, HP-210 Victoreen Radector III Technical Associates PRS-3, Staplex
Smear Counter:	Eberline Sac-4 NMC Proportional Counter Model PCC 11T
General:	Eberline PRS-1 Nuclear-Chicago Model 8775

Confirmatory Survey Results

The NRC requested that Oak Ridge Associated Universities (ORAU) perform a radiological survey of the Zion Nuclear Training Reactor. The survey was conducted on June 9, and 10, 1988, and included surface scans and measurements to determine alpha and beta-gamma activity, smears to determine removable activity levels, exposure rate measurements, and collection of gamma spectra. Results of the survey confirmed that total and removable surface activity and exposure rates in the facility met the applicable criteria.

Future use of Facility

There is no information concerning the proposed future use of the facility in the Docket File.

5. License Termination Date

The NRC approved license termination on October 27, 1988.

6. Federal Register Notice

The Order Terminating Facility License R-119 was published in the Federal Register on October 27, 1988.

7. Hazardous Waste Identified at the Site

No hazardous wastes were indicated in the Docket File.

8. Restriction/Conditions/Concerns

No restrictions, conditions, or concerns were noted in the Docket File.

9. Does the Site meet Present Criteria?

Yes, the information provided from the surveys performed by the licensee and ORAU indicate that the site does meet present criteria.

10. Recommendations

A copy of the ORAU Confirmatory Radiological Survey Report should be included in the Docket File.

Prepared By: A.T. Payne
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: February 19, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Rockwell International Corporation
Address:	Canoga Park, California
Facility:	Solution Type Research Reactor (L-77)
Site Contact:	M.E. Remley, Director Health, Safety, and Radiation
NRC contact:	K.R. Goller, Office of Nuclear Reactor Regulation
Docket No.:	50-94
License No.:	R-40

1. Summary of Operating History

License R-40 was approved on May 17, 1958. It allowed the possession and use of up to 1.5 Kg contained U-235, and up to 12 grams of plutonium contained in a Pu-Be source. The license applied to the L-77 reactor located in building 004 of the Canoga Park facility. The reactor was a Water Boiler type with an operational power limit of 10 watts thermal. The fuel was 19.94% enriched uranyl sulfate dissolved in water and contained in a spherical stainless steel core. The actual inventory on site was 1408 grams of U-235 and 11.86 grams plutonium.

The supporting programs associated with the L-77 reactor were transferred to another facility on July 31, 1974. There is no specific description of the types of research programs that were conducted. The reactor was shutdown on September 30, 1974. In a September 28, 1976, letter the licensee reported that the fuel solution was removed from the reactor and stored in shielded containers in the reactor facility. The fuel was eventually sold to West Germany. License R-40 was terminated on February 11, 1982.

Future Use of Facility

The docket file contains no information in the future use of the facility. However, the termination order states that the area was available for unrestricted use.

2. Request For Termination of License

The request to dismantle reactor L-77 and terminate license R-40 was submitted on January 28, 1976. The reactor was to be dismantled into its component parts, decontaminated, and released for use in other Atomics International projects. Items not meeting the release criteria were to be disposed of at licensed burial sites or transferred to other licenses. The plan stated that the facility would be surveyed after all components were removed. No survey plan was included.

3. Dismantling Order Date

The dismantling order was issued on September 29, 1976.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

There was no specific information on the radiological status of the facility prior to decommissioning other than information on licensed quantities.

Applicable Guidelines, Supplemental Limits, Exceptions

The surface contamination levels listed in Table 1 of Regulatory Guide 1.86 were used as the decontamination criteria. Exposure rate surveys for induced activity were also performed although no specific release criteria was mentioned.

Licensee's Survey

The licensee submitted the "Radiation Survey Report of the L-77 Reactor Facility Following Dismantlement and Decontamination of the Facility" on January 6, 1982. Samples of the shield tank water indicated less than 3E-08 μCi/ml beta activity and no alpha activity above the detection level of 3E-10 μCi/ml. All items removed from the reactor facility were surveyed for total and removable activity. No activity "above background" was detected.

Surface alpha activity was reported as less than 100 dpm/100 cm^2. No specific direct survey result for beta-gamma radiation was mentioned. However, the report states that the facility was checked for "induced radiation" using a thin-window pancake GM probe (among other instruments) with "no residual activity detected above background".

Over 1000 smear samples for removable activity were performed. The samples were analyzed for both alpha and beta. There was no specific documentation on sample locations or results by location. Samples were collected from walls, ceiling, floors, and overhead pipework. Results were reported as "all smear tests indicated less than 20 dpm/100 cm^2 alpha and less than 50 dpm/100 cm^2 beta".

No information on survey sample location or number was listed in the report although maps of the facilities surveyed were included.

The instruments used were described but no calibration information was included. Limited measurement, sampling and analysis procedures were included.

The conclusion of the report states that the results of all radiation surveys are well under acceptable surface contamination levels for the nuclides listed in Table 1 of Regulatory Guide 1.86.

Confirmatory Survey Results

A confirmatory survey was conducted by E.M. Garcia, Region V, on February 12,

1982. The inspection included a review of records, interviews with personnel, and independent surveys. The inspection report stated that the inspection confirmed the licenses findings presented in the January 6, 1982, post dismantling survey report. The inspectors review of the licensees survey records indicated that the reactor facility was divided into approximately 1 square meter sections with as many as six smear samples collected in each section. The inspector verified that the instruments used by the licensee were within specified calibration dates and that the calibrations were traceable to NBS.

The inspector performed surveys of all three rooms of the facility for removable and fixed or induced contamination. All 48 smear results were less than 20 dpm/100 cm^2 alpha and less than 50 dpm/100 cm^2 beta-gamma. Direct radiation measurements identified "no residual activity above background."

All instruments except the alpha probe were listed with serial numbers and calibration dates. No measurement, analysis, or sampling procedures were included in the report.

5. License Termination Date

The termination order was issued on February 11, 1982.

6. Federal Register Notice

The order terminating the facility license was transmitted for publication February 23, 1982.

7. Hazardous Waste identified at the Site

No hazardous wastes were identified in the docket file.

8. Restrictions/Conditions/Concerns

No restrictions were noted in the docket file.

9. Does the Site Meet the Current Criteria?

The licensee's post dismantling survey report indicates that the surface contamination levels listed in Regulatory Guide 1.86 were met. There is no specific reference to exposure rates being less than 5 μR/hr above background although exposure rate measurements were made with appropriate, calibrated instruments, and reported as less than background. All of the licensees results were verified by the NRC's confirmatory survey. Specific sample locations and results were not listed in either the licensees or the NRC's report.

10. Recommendations

The survey results should have been presented in a more comprehensive, explicit format allowing each survey result to be reviewed. However the Regulatory Guide 1.86 criteria were specifically stated as having been met. The 5 μR/hr above background exposure rate criteria were not specifically mentioned although it appears that the criteria were met. A limited exposure rate survey should be performed to fully document that the site meets the criteria.

Prepared by: David N. Fauver
Division of Low-Level Waste Management and Decommissioning
Office of Nuclear Material Safety and Safeguards

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	University of Delaware
Address:	Newark, Delaware
Facility:	AGN 201 Training Reactor
Site Contact:	I.G. Greenfield, Reactor Administrator
NRC contact:	K. Goller, Assistant Director for Operating Reactors
Docket No.:	50-98
License No.:	R-43

1. Summa.; of Operating History

The University of Delaware was licensed on July 3, 1958, to operate the AGN 201 training reactor at power levels up to 0.1 watt. The fuel was a mixture of polyethylene and uranium oxide enriched to 20% U-235. The reactor was used to conduct research and development activities.

Between startup on July 12, 1958, and the final reactor shutdown on November 20, 1974, the total power output was 49.3 watt-hrs. There is no documentation indicating any release or spills of radioactive material or burial under 10 CFR 20.304 The reactor was designed as a completely sealed system, and in normal operation as well as during disassembly activity no gaseous or liquid effluent was planned. On December 14, 1957, the fuel was removed from the reactor and shipped to Oak Ridge. License R-43 was terminated on February 26, 1979.

Future Use of Facility

The rooms housing the AGN 201 reactor were to be released for unrestricted use and modified for other university activities.

2. Request For Termination of License

On January 28, 1978, the licensee requested authorization to dismantle the reactor according to the dismantling plan submitted on July 27, 1977, and that the NRC terminate license R-43. The plan called for the removal of the Pu-Be start-up source and control rods. The Pu-Be source was transferred to license SNM-656. The fission plate, fuel sections of the control rods and fuel assemblies were shipped to Oak Ridge. All contaminated material was to be disposed of as rad-waste. The plan states that no transferred were to be released through cold drains until shown to be within applicable limits.

3. Dismantling Order Date

The dismantling order was issued on June 12, 1978.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The license authorized the possession and use of up to 700 grams of contained U-235 and 16 grams of plutonium contained in a Pu-Be source. Calculations by the licensee indicated that the fission product activity in the core was 2 orders of magnitude less than the U-235 activity. This was due to the reactors low power output and the approximately 3 year decay time between the time the reactor was shutdown and the time it was decommissioned. Therefore U-235 was considered the primary contaminant.

Applicable Guidelines, Supplemental Limits, Exceptions

The surface contamination limits in Regulatory Guide 1.86 were used as the criteria for unrestricted release of the facility.
Licensee's Survey

The licensee's post dismantling survey results were submitted in the report "Defueling and Disposal of the University of Delaware AGN-201 Training Reactor" on January 18, 1978.

The water drained from the reflector tank was sampled before release to the sanitary sewer system. Analysis with a liquid scintillation counter resulted in 0 cpm above background. External gamma surveys of the loaded fuel shipping casks were less than 0.1 mR/hr. The control console, miscellaneous components, and main reactor tank were surveyed for direct and removable radiation. All results were reported as 0 dpm.

The reactor rooms were surveyed for direct and removable surface contamination two days after the fuel was removed. Followup measurements were performed 13 days later. These surveys included the reactor room floor directly underneath the reactor and surrounding areas, door knobs, table tops, and bench tops in the reactor room, room 146 and room 246. All measurement results were reported as 0 dpm.

The shipping casks were surveyed for gamma radiation. The results were reported as less than 0.1 mR/hr. No gamma surveys of the facility was reported.

The sampling, measurement, and analysis procedures were documented in the licensees survey report including a complete listing of the instruments used and calibration information. Each result was reported by survey location. The report contains no exposure rates at 1 meter from facility surfaces.

On October 12, 1978, the licensee reported that all of the AGN-201 reactor components had been removed. The previous survey was performed with the reactor tank in place in the reactor room. Surveys for removable contamination in the reactor room were performed. All results were reported as background. No information on the measurement , sampling, and analysis technique was provided.

Confirmatory Survey Results

The closeout inspection was performed by K.E. Plumlee, Region I, on December 11, 1978. The inspection included confirmatory surveys and a review of licensee survey, dismantling, fuel transfer,and component disposal records. No problems or discrepancies in the licensees records were noted.

The confirmatory survey included swipes and surveys with portable survey instruments. The measurement, sampling, and analytical procedures were outlined in the inspection report. Swipe samples were obtained from the reactor floor area, floor drain cover, and residue inside the floor drain. The inspector made direct surface contamination surveys of the training center, corridors, a fire lane, and each storage area, laboratory, and room adjoining the training center. The inspector reported "no detectable radiation".

The direct contamination survey detection limit for alpha and beta was reported to be 1000 dpm/100 cm^2. The removable contamination survey detection limit was 10 dpm/100 cm^2 for alpha and beta-gamma.

No area exposure rates surveys were performed.

Based on the review of the licensees records and confirmatory surveys the inspector concluded that the AGN-201 facility met the surface contamination criteria of Regulatory Guide 1.86 and could be released for unrestricted use.

5. License Termination Date

The order terminating license R-43 was issued on February 26, 1979.

6. Federal Register Notice

The license termination order was submitted for publication on March 6, 1979.

7. Hazardous Waste identified at the Site

No release of hazardous waste was noted in the docket file.

8. Restrictions/Conditions/Concerns

No restrictions or concerns were noted in the docket.

9. Does the Site Meet the Current Criteria?

The licensee and NRC surveys indicate that the AGN-201 facility met the unrestricted release criteria of Regulatory Guide 1.86. There was no information on the exposure rates at 1 meter from facility surfaces therefore the 5μR/hr survey criteria cannot be evaluated.

10. Recommendations

Limited surveys should be performed to confirm that the facility meets the 5 μR/hr exposure rate release criteria.

Prepared by: David N. Fauver
Division of Low-Level Waste Management and Decommissioning
Office of Nuclear Material Safety and Safeguards

REVIEW OF TEST AND RESEARCH REACTOR TERMINATED LICENSES

Site:	Babcock & Wilcox Lynchburg Research Center
Address:	P.O. Box 229, Lynchburg, Virginia 24505
Facility:	Lynchburg Pool Reactor
Site Contact:	A.F. Olsen, Senior License Administrator, Research and Development Division
NRC Contact:	P.B. Erickson, Division of Licensing
Docket No.:	50-99
License No:	R-47

1. Summary of Operating History

Lynchburg Research Center (LRC) is located in Campbell County, Virginia approximately three miles from Lynchburg, Virginia. From September 19, 1958 through September 16, 1981, Babcock & Wilcox (B&W) operated the Lynchburg Pool Reactor (LPR). The LPR facility was utilized for instrumentation development, neutron radiography, operator training, neutron activation analysis, and neutron transmission studies. The LPR was a one megawatt (MW) "swimming pool" type reactor. The fuel consisted of plate-type elements enriched with U-235. The reactor operated intermittently since 1958 and generated a total of 35 MW days of thermal energy.

LPR operations were supported by several counting rooms, a large hot cell complex, radiochemistry laboratories, machine shops, electronic maintenance laboratories, and personnel dosimetry calibration facilities.

In 1972, the LPR pool was emptied to repair a leak and radiation levels were measured when the pool was empty. The maximum radiation level was 2 R/h at 1 foot which was due to the control rods. The next highest level was 100 mR/h at 1 foot from the beam ports. These low levels were expected since most of the structures near the core were aluminum alloys which retain very little activation.

All fuel was removed from the facility and shipped to the Department of Energy's Savannah River plant for reprocessing. The remaining radioactivity consisted of the activated and/or contaminated components, concrete, piping, heat exchangers, demineralizers, resins, experimental capsules, and start up sources. Most materials near the core were aluminum 1100 or 6061. The most important of the alloy elements was Zn-65 and the most important of the impurities was Co-60. These two isotopes were expected to comprise the bulk of the radioisotopes after the fuel was removed, with a total curie content expected to be less than one curie.

The reactor water was contaminated to levels near or below that acceptable for release to uncontrolled areas (10 CFR Part 20).

Radioactive liquids were collected in underground storage tanks at the liquid waste disposal building. When a tank was full, it was agitated and sampled. It was released to the B&W Naval Nuclear Fuel Division liquid waste treatment center if activity was less than 25% of the MPC values specified in 10 CFR 20, Appendix B, Table 1, Column 2. If the activity in the tank exceeded these guidelines, dilution was used to meet the limits.

2. Request for Termination of the License

A request for termination of License R-47 was submitted to the NRC on April 23, 1982.

3. Dismantling Order Date

The NRC approved dismantling and signed the Dismantling Order on January 11, 1982.

The decommissioning plan was prepared by B&W and submitted to the NRC in July 1981. Dismantlement operations were planned to consist of removal of the control rods and rod drives, experimental assemblies, graphite reflector elements and control rod poison elements prior to draining the pool. Dismantling would also consist of removal of activated reactor internal components, beam ports, and autoclave. Dismantlement was also expected to involve the removal of significant quantities of activated concrete from the lower pool area to reduce the radiation to levels acceptable for unrestricted access. B&W planned to decontaminate the facility and if they were unable to meet the release criteria, they would request that residual activity after dismantlement be included in the LRC Special Nuclear Materials (SNM) license (SNM-778).

The plan was to leave all structures on the site standing with very little change, with the exception of the nonradioactive redwood cooling tower which would be removed. Following dismantlement, the LPR building could then be put to use in other programs at the LRC. Nonradioactive components would be put into use in other programs or disposed of as scrap. Radioactive sources and reactor parts would be disposed of in an authorized burial ground or retained for use under the SNM-778 license.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

After removal of fuel from the facility, the remaining radioactivity consisted of the activated and/or contaminated components, concrete, piping, heat exchangers, demineralizers, resins, experimental capsules, and start up sources. Most material near the core was aluminum 1100 or 6061. Hence, the most important of the alloy elements was Zn-65 and the most important of the impurities was Co-60. These two isotopes were expected to comprise the bulk of the radioisotopes after the fuel was removed, with a total curie content expected to be less than one curie. The reactor water was contaminated at levels near or below that acceptable for release to uncontrolled areas (10 CFR 20).

NRC inspection report 50-99/82-02 performed on April 19, 1982 noted that the remaining radioactivity in the LPR consisted of activation products (Eu-152) in the concrete pool walls and metal activation products (primarily Co-60), the majority of which was located in the 8" beam port on the interior end of the ports.

Applicable Guidelines, Supplemental Limits, Exceptions

In accordance with the Dismantling Plan, surfaces would be decontaminated to levels consistent with Table 1 of Regulatory Guide 1.86. Other radioactive material such as Co-60, Eu-152, and Cs-137 that may exist in concrete, components, structures and soil would be removed such that the radiation levels from these isotopes was less than 5 μR/h above natural background as measured at one meter from the surface. However, B&W did not plan to release the facility for unrestricted access, but was planning on transferring the entire area to the control of the SNM-778 license of the LRC.

Licensee's Survey

In April, the Licensee performed radiation exposure rate measurements at one meter above the floor in the heat exchanger room, pool, and along one edge of the pool reactor basement. Smear surveys were also performed on the floors and lower walls (three foot above the floor) in the heat exchanger room, pool, pool reactor basement, upper level pool area, storage deck, control room, and staircases.

Measurement, Sampling, and Analyses Procedures

The Dismantling Plan stated that all decommissioning activities would be conducted in accordance with the Dismantling Plan. Detailed procedures implementing this plan would be in writing and approved by the Operations Supervisor and Safety Review Committee of B&W.

No procedures were presented or discussed in the Docket File.

Final Survey Data

Exposure rate measurements were performed using a Reuter-Stokes Model RSS-111 pressurized ion chamber. The exposure rate measurements performed indicated that the exposure rates ranged from 9 TO 13 μR/h, with a background of 9 μR/h.

The smear survey performed indicated residual contamination ranging from 78 to 92 dpm/100 cm^2, with a counter background of 78 dpm.

Confirmatory Survey Results

In March, the NRC collected samples from the north and east walls of the facility at depths ranging from 0 - 15 inches. Smears were also taken in the control room, upper pool area, storage deck, office, heat exchanger room, pool and basement. A water sample was obtained from the LPR waste tank. Air samples were obtained while various tasks were being performed and soil samples were obtained from the pool floor.

Gamma spectroscopy was performed on the samples collected by the NRC in March. The results of core drilling indicted Co-60 at 2.58 E+01 dpm/g, Eu-152 at 9.5 E+01 dpm/g, Co-57 at 2.12 E+01 dpm/g, and Na-22 at 2.3 dpm/g. The sample from the LPR waste tank indicated Co-57 at <9.6 E-08 dpm/cc and Co-60 at <2.3 E-07 dpm/cc. The air filter indicated Co-57 at 2.7 E+01 dpm/sample and Co-60 at 2.0 E+01 dpm/sample. The dirt from the pool floor indicated Co-57 at 2.18 dpm/g, Co-60 at 4.0 dpm/g, and Eu-152 at 5.9 dpm/g.

The smears indicated Co-57 at < 1.0 E-06 dpm total, Co-60 at < 2.4 E-06 dpm total, and Eu-152 at < 1.7 E-05 dpm total.

In May, the NRC also performed radiation measurements using an Eberline PRM-7 in the heat exchanger room, pool and along one edge of the pool reactor basement. The measurements ranged from 9.5 to 12 μR/h, with a background of 9 μR/h. Smear surveys were also performed in the heat exchanger room, pool, pool reactor basement, upper level pool area, storage deck, control room, and staircases. The results of NRC smear surveys are not presented in the Docket File, but the report indicates that the results confirmed the Licensee's Final Survey and that levels were within the specified guidelines.

Future Use of Facility

Following dismantlement, the LPR building would be put to use in other programs at the Lynchburg Research Center. Non-radioactive components would be put to use at the LRC for other projects or disposed of as scrap. Radioactive sources and reactor parts would be disposed of in an authorized burial ground or retained for use under the SNM-778 license.

5. License Termination Date

On July 20, 1982 the NRC found that the facility had been dismantled and decontaminated, and that satisfactory disposition had been made of components, parts and fuel, and approved termination of the license.

6. Federal Register Notice

No Federal Register notices are included in the Docket File.

7. Hazardous Waste Identified at the Site

Yes. The system in the autoclave area was covered with asbestos.

8. Restrictions/Conditions/Concerns

No restrictions, conditions, or concerns were indicated in the Docket File.

9. Does the Site Meet Present Criteria?

The surveys performed by the Licensee and NRC indicate that certain rooms of the LPR meet the present criteria for release for unrestricted access.

The Docket File does not contain specific procedures or survey results other than those noted in this report. The only areas surveyed were the heat exchanger room, pool, pool reactor basement, upper pool area, storage deck, control room and staircases; but there were no radiation exposure measurements performed by the Licensee in the pool reactor basement, storage deck, control room, office, stairs or roof and smears were not taken in the office or roof. Additionally the Licensee did not address the upper walls and ceilings. No other areas were surveyed such as the pump room, offices, laboratories, counting rooms, roof, etc. Also, direct measurements for residual contamination were not performed.

10. Recommendations

Unless further survey information is available, it may be necessary to perform a more detailed survey to provide assurance that all areas of the facility have been decontaminated to meet the release criteria. However, due to the time elapsed since the termination of operations and other uses that may have occurred since license termination, it is unlikely that residual contamination from the R-47 license would still be present. Furthermore, even if any contamination were found, it would be very difficult to associate it with the license.

Prepared by: Betty M. Smith
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: January 29, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Gulf United Nuclear Fuels
Address:	Pawling, New York
Facility:	Pawling Lattice Test Rig Facility (PLATR)
Site Contact:	P. Loyson, Regulatory Administration
NRC contact:	D.J. Skovholt, Directorate of Licensing
Docket No.:	50-101
License No.:	R-49

1. Summary of Operating History

The PLATR facility was licensed in October 22, 1958, to operate at 5 watts thermal (100 watt thermal for short periods of time with NRC approval). The reactor was used to verify the validity of computer codes for nuclear fuel management. The docket file contains no information on the reactor design or operating history other than a statement that the reactor was operated intermittently at power levels of a few watts. The reactor was officially deactivated on August 9, 1973, although the records indicate that it was shutdown sometime before that with no specific date mentioned. There was no record of any unplanned release or spill or burial under 10 CFR 20.304. License R-49 was terminated June 25, 1974.

Future Use of Facility

The docket file contains no information on the future use of the facility.

2. Request For Termination of License

The request for license termination was submitted on October 28, 1971. At this time the building, and more specifically the room, in which the assembly vessel was located also contained the Proof Test Facility (PTF) which was operated under AEC license CX-25 and remained under direct control of the licensee.

An application for a possession only license was filed June 8, 1973. At this time the fuel had been removed and the reactor dismantled.

On October 23, 1973, a second request for license termination was filed in conjunction with the termination request for the PTF facility (license CX-25). All correspondence after October 23, 1973, referred to both the R-49 and CX-25 license. The two facilities were decommissioned simultaneously.

3. Dismantling Order Date

The dismantling order was issued January 25, 1974.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

According to a November 6, 1972, letter from the licensee the material under license R-49 included 3,261 Kg of uranium (2.4% U-235) and 4.0 Kg Plutonium. No other nuclides were referenced in the docket file.

Applicable Guidelines, Supplemental Limits, Exceptions

The NRC closeout survey referenced the "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of License for Byproduct, Source or Special Nuclear Material." Limits referenced in other parts of the closeout survey report indicate that the April 22, 1970 version was used.

Licensee's Survey

A preliminary smear survey of the PLATR components and the reactor room walls and floor, conducted October 26, 1971, resulted a maximum of 5 dpm alpha and 10 dpm beta-gamma. The sample locations were reported but the smear sample area was not. "Instrument survey" results were reported as less than 1 mR/hr.

A pre-decontamination survey, using a Texas Nuclear Model 2650 survey meter, identified 0.01-0.05 mR/hr in the "PLATR area". Background areas were 0.02-0.05 mR/hr. Contamination surveys of the PLATR components resulted in 0-8 dpm/100 cm^2 alpha and 0 - 18 dpm/100 cm^2 beta-gamma.The contamination levels of the rooms comprising the facility were 0-7 dpm/100 cm^2 alpha and 0-5 dpm/cm^2 beta-gamma. The reports does not clarify if the contamination surveys were direct or removable. The sample locations are reported generally with no maps or diagrams included. Sampling and analysis procedures are not reported.

A post decontamination survey was performed (sometime between January 25 and March 13, 1974). All instruments used for the survey were listed. No calibration information or description of measurement, sampling, or analysis procedures were provided. Over 200 survey results were listed by location. Maps and drawings were not included. The direct contamination survey results were reported as less than 200 dpm/100 cm^2 alpha and less than 0.02 mR/hr beta-gamma for both the facility surfaces and the components. Removable activity on components ranged from 0-6 alpha and 2-18 beta-gamma dpm/100 cm^2; on building surfaces the removable levels were 0-22 alpha and 2-31 beta-gamma dpm/100 cm^2.

Confirmatory Survey Results

The NRC closeout inspection was conducted by P.C. Jerman, Region I, on April 9-11 and 15-16, 1974. The measurement, sampling and analysis procedures were presented in the report. All instruments were listed along with serial numbers and calibration dates. The results of a "detailed" survey of the critical facility showed "no contamination exceeding the guidelines."

5. License Termination Date

   License 50-101 was terminated on June 25, 1974.

6. Federal Register Notice

   The termination order was transmitted for publication in the Federal Register on July 3, 1974.

7. Hazardous Waste identified at the Site

   No hazardous waste was identified in the docket file.

8. Restrictions/Conditions/Concerns

   No restrictions or conditions were indicated in the docket file.

9. Does the Site Meet the Current Criteria?

   The sensitivity of the licensees direct alpha surveys was 200 dpm/100 cm^2. This is greater than the direct contamination limit for plutonium in Regulatory Guide 1.86 therefore an evaluation of this criteria cannot be made. Also, no exposure rate measurements were performed.

   The critical facility which housed the PLATR facility was surveyed in 1986 by Oak Ridge Associated Universities (Report # ORAU 88/G-89) at the request of the current owner; the Department of the Interior.The findings of the survey demonstrated that the radiological status of the facility satisfied the current surface contamination release criteria for unrestricted release. The Oak Ridge survey results do not include exposure rate measurements to allow comparisons with the criteria of 5 μR/hr above background at 1 meter.

10. Recommendations

A limited exposure rate survey at 1 meter from the facility surfaces should be performed to complete the survey documentation.

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REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Oregon State University
Address:	Corvallis, Oregon
Facility:	AGN 201 Reactor
Site Contact:	C.H. Wang, Reactor Administrator
NRC contact:	R.W. Reid, Operating Reactors Branch
Docket No.:	50-106
License No.:	R-51

1. Summary of Operating History

The AGN 201 reactor at Oregon State was licensed to operate at a maximum power level of 0.1 watt. The fuel was comprised of polyethylene and 20% enriched uranium oxide. The reactor was used to conduct educational activities. It first went critical on January 28, 1959. The AGN 201 reactor was designed as a completely sealed system, and in normal operation as well as during disassembly activity no gaseous or liquid effluent was planned.

The reactor was used continuously until December 1974 at which time the reactor was shut down. The total power output of the reactor over it's 16 years of operation was 1.1 watt-day. The docket file contains no record of an unplanned release or spill of radioactive material or disposal under 10 CFR 20.302 or 20.304. License R-51 was terminated November 10, 1981.

Future Use of Facility

The docket file contains no information on the future use of the facility. The termination order stated that the area was available for unrestricted use.

2. Request For Termination of License

The licensee submitted a dismantling and disposal plan December 18, 1978. The core and all control rods were removed from the "AGN" room and stored in the TRIGA reactor room at Oregon State. The special nuclear material in these components was subsequently transferred to the TRIGA license, No. R-106. The remaining reactor components were cleared for unrestricted release and moved to an Oregon State storage facility. The 10 mCi Ra-Be startup source was transferred to Oregon materials license ORE-0005-3.

3. Dismantling Order Date

The dismantling order was issued March 8, 1979.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

License R-51 authorized the possession and use of 700 grams of contained U-235. The dismantling plan contains a calculation that estimated the fission product activity in the core at the time of decommissioning to be 48.9 μCi. This was approximately 50 times less than the activity of the uranium in the core. Therefore, the primary contaminant was assumed to be uranium.

Applicable Guidelines, Supplemental Limits, Exceptions

Regulatory Guide 1.86 was the used to determine if the site could be released for unrestricted use.

Licensee's Survey

The licensees final decommissioning report was submitted July 9, 1980. The report summarized the final survey results but did not include specific survey location or results. The licensee stated in this report that all of the survey results were on file for NRC review. A blank survey form is included as an appendix to demonstrate the information collected during the surveys. The survey form appears to include all of the information necessary to evaluate the survey results including area surveyed, instrument used, instrument background, and proper units.

The final area survey included direct surface contamination surveys of all major objects and room surfaces, including the entire floor, all cabinets, all work surfaces, book cases, furniture, and walls. The survey produced "no detectable radiation levels above background values". Smears of "major objects and surfaces" were also performed with "no radioactivity above background" detected.

No exposure rate surveys were performed.

Confirmatory Survey Results

The closeout inspection was conducted on September 9, 1980, by J.R. Curtis, Region V. The inspector toured the facility, conducted meter and smear surveys, and examined licensee survey records.

The inspection report stated that the examination of the licensees survey records and discussions with the licensees staff who participated in dismantling operation disclosed that no radiation levels or radioactivity above normal background levels were detected on the floor surfaces of the facility.

The measurement, sampling, and analysis procedures used during the inspectors confirmatory survey were documented and complete although specific sample locations were not identified. Gamma radiation levels were measured with a Micro-R meter at approximately three feet above the floor throughout the facility and within six inches of the floor where the reactor was located. The report stated that "no radiation above the 5-15 μR/hr background level was identified".

Removable and direct surface contamination surveys were conducted at "selected locations" in the facility with no result "above background." The inspectors conclusion was that the facility met the unrestricted release criteria of Regulatory Guide 1.86.

5. License Termination Date

License R-51 was terminated on November 10, 1981

6. Federal Register Notice

The order terminating the facility license was transmitted for publication in the Federal Register on November 12, 1981

7. Hazardous Waste identified at the Site

No release of hazardous waste was identified in the docket file

8. Restrictions/Conditions/Concerns

There were no restrictions or concerns noted in the docket file.

9. Does the Site Meet the Current Criteria?

Oregon State Universities final survey indicates that the reactor site meets the surface contamination criteria for unrestricted release in Regulatory Guide 1.86. The licensee did not perform exposure rate surveys of the facility.

The NRC confirmatory survey does provide results indicating that the exposure rate, measured six inches from the floor area where the reactor was located, was below the 5 μR/hr above background limit.

10. Recommendations

The licenses survey data should be included in the docket file for completeness. However, the inspector stated that he reviewed the survey data and agreed with the summary of the results in the licensee's final survey report.

The information provided in the docket file is adequate to conclude that the facility meets the current release criteria.

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REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Allis-Chalmers Manufacturing Company
Address:	Greendale, Wisconsin
Facility:	Critical Experiment Facility
Site Contact:	T.D. Lyons
NRC contact:	D.L. Doan, Division of Reactor Licensing
Docket No.:	50-108
License No.:	CX-15

1. Summary of Operating History

The critical experiment facility was licensed on November 3, 1959, to operate at power levels up to 100 watts thermal. No information on reactor design or operating history was included in the docket file. There were no documented unplanned releases or spills of radioactive material or burials under 10 CFR 20.304. The last experiment was performed on July 20, 1962.

Future Use of Facility

There was no information on the future use of the facility.

2. Request For Termination of License

The licensee requested that license CX-15 be terminated on September 12, 1966. The letter stated that the fuel pins and major core components had been shipped to General Electric at Vallecitos Atomic Laboratory. At the request of the NRC, additional information was supplied on October 18, 1966.

The dismantling plan stated that reactor components with "less than 50 dpm of fixed alpha activity,less than 10 dpm of loose alpha activity, and less than 0.2 mR/hr beta-gamma activity" would be released for unrestricted use. The licensee committed to dispose of all solid waste in accordance with AEC regulations and survey each critical facility component as it became accessible. Liquid waste was to be evaporated and handled as radioactive waste.

3. Dismantling Order Date

The dismantling order was issued December 8, 1966.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

No information was provided on the primary contaminant at the facility. The operating license authorized the possession and use of 270.1 Kg of contained U-235.

Applicable Guidelines, Supplemental Limits, Exceptions

The criteria in "Radioactivity Contamination Limits for Abandonment of Facilities and Equipment" was used to determine acceptability for unrestricted release.

Licensee's Survey

The October 18, 1966, supplement to the dismantling plan include pre-dismantling surveys. The instruments used were described in the report. Limited measurement sampling and analysis procedures were also included.

Points at approximately six foot distances on the walls and floor of the compartment housing the critical facility were surveyed for fixed and removable contamination. The highest fixed beta-gamma activity was 0.1 mR/hr. The same locations were surveyed for removable contamination. The results showed "no detectable activity on any of the surface areas". The report is unclear as to whether the smear samples were analyzed for beta-gamma, alpha, or both. No fixed alpha results were reported.

All accessible areas of the critical facility were surveyed. A high reading of 0.5 mR/hr beta-gamma was obtained by lowering the probe directly into the reactor vessel. Smear samples in the vessel "showed no detectable activity". Water samples from the reactor dump tanks contained "no detectable" alpha or beta-gamma activity and was released to the sewer.

The licensees post dismantling survey was submitted on October 2, 1967. The description of instruments, and measurement procedures in the dismantling plan were referenced. No additional information was provided.The survey included the components removed during the dismantling and the floors and walls of the reactor room. Two items were identified as exceeding the release criteria defined in the dismantling plan: a tube through which the source was introduced read 1.0 mR/hr and a tar-like substance in an expansion joint of the concrete floor read 75 mR/hr. The tube was cut up and the tar-like substance removed. Both were disposed of as solid radioactive waste. The results of the floor and wall surveys were reported as less than the limits listed in the dismantling plan.

Confirmatory Survey Results

A closeout survey was conducted October 6, 1967, by C.D. Hampleman, Region III. The inspector reported that all equipment had been removed from the room which housed the CX-11 facility and disposed of as salvage or radioactive waste. Contaminated waste was shipped to Nuclear Engineering Company. No radioactive materials remained at the Greenville plant. The inspection included the review of "several hundred" licensee surveys of equipment, walls, and floors. All of the surveys indicated "no activity detected."

The inspector performed direct alpha and beta-gamma surveys of all accessible surfaces. At that time the room consisted of bare concrete walls and floors. The report stated that "no radioactivity above background was noted." Smears were collected from the floors and walls and analyzed for alpha and beta-gamma by Argonne National Laboratory. No results "above background" were reported.

The instruments used for the confirmatory survey were listed. However no calibration information or measurement, sampling, or analysis procedures were included.

5. License Termination Date

License CX-15 was terminated on November 20, 1967.

6. Federal Register Notice

The Notice Of Termination Of Facility License was submitted for publication on November 2, 1967.

7. Hazardous Waste identified at the Site

No hazardous waste was identified in the Docket file.

8. Restrictions/Conditions/Concerns

No restrictions were noted in the docket file.

9. Does the Site Meet the Current Criteria?

The current residual direct and removable alpha contamination criteria appears to have been met although no specific sampling locations were reported. The fixed beta-gamma evaluation criteria listed in the licensees dismantling report, i.e., 0.02 mR/hr, cannot be compared to the current criteria. It is not clear whether the smear samples were analyzed for removable beta-gamma activity. No exposure rate surveys were listed therefore the 5 μR/hr above background criteria cannot be evaluated.

10. Recommendations

A limited survey should be performed at the CX-~~11~~ 15 facility if it still exists.

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Site:	North Carolina State
Address:	Raleigh, North Carolina
Facility:	R-3 Training and Research Reactor (NCSUR-3)
Site Contact:	R.G. Cockrell, Director Nuclear Reactor program
NRC contact:	D.M. Crutchfield, Operating Reactors
Docket No.:	50-111
License No.:	R-63

1. Summary of Operating History

The NCSUR-3 was licensed on March 16, 1960, to operate at a maximum power level of 10 kilowatts. The reactor was a graphite reflected tank-type reactor using 18 plate MTR-type fuel elements. The NCSUR-3 reactor was operated from 1960-1973. The total power output over its operating lifetime was 52.5 MW-hrs. No unplanned releases or spills, or disposals under 10 CFR 20.304 or 20.302 were documented.

In 1973 the core was unloaded, the fuel placed in storage racks within the reactor tank, the control rods moved to storage, and the electrical controls disconnected. In February 1974, the fuel was placed in dry storage and the reactor system drained. The top of the reactor was covered by a two inch steel plate. The termination order was issued June 13, 1983.

Future Use of Facility

The Burlington Nuclear Laboratory building, which housed the reactor, was to be renovated or demolished. The dismantling plan called for the release of the facility for unrestricted use.

2. Request For Termination of License

The dismantling plan was submitted June 5, 1980. The plan detailed the dismantling operation, as well as the reactor operating history, the current site status, safety aspects, and radiological protection considerations. A final survey was to be performed but no details on the measurement, sampling, or analysis procedures were included.On February 19, 1981, a formal request to terminate the reactor was submitted. The irradiated fuel was shipped to DOE at Savannah River. Remaining unirradiated fuel and the PuBe start up source were transferred to License R-120. Dismantling began in the fall 1981. The demolition and removal of the concrete bioshield was scheduled for completion in January 1983.

3. Dismantling Order Date

The dismantling order was issued on June 1, 1981.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

License R-63 authorized the possession and use of 4.02 kg of contained U-235 and 80 g of Plutonium for use in a PuBe start up source. The dismantling plan stated that the primary isotope expected during decommissioning was Eu-152 from the activation of concrete. Co-60 and Ba-133 were also identified.

Applicable Guidelines, Supplemental Limits, Exceptions

An addendum to the dismantling plan dated April 30, 1981, stated that Regulatory Guide 1.86 would be used as the release criteria for surface contamination. The limit for residual activation and fission products in reactor components and concrete was 5 μR/hr above background at 1 meter.

Licensee's Survey

The licensee's final survey was submitted March 29, 1983. The cover letter reported removable beta-gamma activity from the walls and floor 82 dpm/100 cm^2 or less. All smears for alpha were reported as 0 dpm. It is assumed that the alpha smears were over a 100 cm^2 area although this is not explicitly stated.

The locations of removable beta-gamma contamination surveys were well documented with maps and descriptions. The location of the alpha smear surveys were not well documented but the report implied that all samples were counted for both alpha and beta-gamma. No measurement or sampling procedures for removable contamination surveys were included in the report. The "Swipe Log" for beta-gamma smears contains limited calculational data including gross cpm, background cpm, efficiency, and net dpm.

No fixed surface contamination surveys were reported.

Exposure rate surveys were performed at 1 meter from the floor of the reactor bay and reactor pit using a Ludlum Micro-R meter. No calibration data or other measurement information was provided. All areas met the 5 μR/hr above background criteria except for an area that was directly under the reactor core which was 7-8 μR/hr above background of 20 μR/hr. The licensee planned to pour 18 inches of concrete over this area which would reduce the exposure rates to background or below. No documentation confirms that this was performed. However, the current exposure rate would be approximately 5 μR/hr without the concrete considering decay (assuming a 13.4 year half-life for Eu-152) during the 8 years since the original survey was performed.

Confirmatory Survey Results

A confirmatory survey was conducted on April 5-6, 1983, by J.B. Kahle, Region II. The proper disposition of the fuel and contaminated material was verified. Surveys of the "hot" spot of 7-8 μR/hr above bkg were made using an Eberline PRM-7. The inspectors results were less than 5 μR/hr above bkg with and without a steel plate covering the spot. The plate was intended to approximate the radiological conditions after the 18 inches of concrete were poured.

Smear surveys for removable contamination were counted for both alpha and beta-gamma. The highest results were 30 dpm/100 cm^2 beta-gamma and 1.9 dpm/100 cm^2 alpha. No direct surface contamination measurements were performed. No measurement, sampling, or analysis procedures were documented other than listing the instrument for the exposure rate survey.

5. License Termination Date

License R-63 was terminated June 13, 1983.

6. Federal Register Notice

The termination order was transmitted for publication in the Federal Register on June 15, 1983.

7. Hazardous Waste identified at the Site

No hazardous wastes were identified at the site.

8. Restrictions/Conditions/Concerns

No restrictions were noted in the docket file

9. Does the Site Meet the Current Criteria?

The site meets the current criteria for removable contamination and exposure rate above background at 1 meter. There is no documentation on direct surface contamination levels.

10. Recommendations

Surveys for direct surface contamination should be performed to assure compliance with current release criteria.

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REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site: University of Oklahoma
Address: Norman, Oklahoma
Facility: AGN 211P
Site Contact: P. Skierkowski, Radiation Safety Officer
NRC contact: T.S. Michaels, Division of Reactor Projects
Docket No.: 50-112
License No.: R-53

1. Summary of Operating History

The AGN211P reactor at the University of Oklahoma was licensed on December 29, 1958, to operate at 15 watts. On November 10, 1983, the maximum power level was raised to 100 watts. The reactor was used for training university students. The maximum reactor power output was approximately 1000 watt-hrs/yr. The AGN 211P was a light water moderated, graphite reflected reactor using less than 20% enriched uranium. The facility was described as consisting of the control panel, tank assembly, east pit, and west pit.

No unplanned releases or spills, or disposals under 10 CFR 20.304 or 20.302 were documented. The reactor was shutdown in April 1986.

Future Use of Facility

The existing reactor area was to be used for radioactive source storage, instrument calibrations and other activities under the universities Broad Form License #35-07466-05.

2. Request For Termination of License

The Dismantling and Decommissioning plan and request to terminate license R-53 were submitted October 25, 1988. The dismantling plan was amended March 9, 1989 to include the current release criteria and other information. The fuel had been removed and shipped to DOE at Oak Ridge prior to the decommissioning request. All components were surveyed as the reactor was dismantled. Activity was detected on the fine control rod (0.1 mR/hr @ 1m) and in an ion exchange resin sample (1.34 pCi/g). The control rod and the resin were disposed of as low level waste. All other components including the reactor superstructure, the fuel storage racks, and irradiation ports were reported as less than background. Water samples collected from the reactor pool and the resin bed were "not greater than background".

3. Dismantling Order Date

The dismantling order was issued on May 26, 1989.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The license authorized the possession and use of 170 g of contained U-235. Cesium 137 was identified in a resin sample. Cobalt 60 was identified as an activation product in a control rod.

Applicable Guidelines, Supplemental Limits, Exceptions

Regulatory Guide 1.86 was used for surface contamination release limits. The limits on exposure rates were 5 μR/hr above background at 1 meter.

Licensee's Survey

The final survey was included in the licensees August 28, 1989, license termination request. Removable surface contamination was surveyed by dividing the tank, pit walls, and floor into a 1 square meter grid. A smear sample was collected in each grid and analyzed for alpha and beta-gamma. All alpha smears results were less than the "alpha plateau background".One sample, on the floor of the east pit, exceeded the beta-gamma background. After cleaning, a second smear " indicated a background level." No direct alpha or beta-gamma surveys were performed.

Exposure rate surveys of the pit and tank walls and the floor were performed at the same grid locations as the removable contamination surveys. Background counts ranged from 43-54 cpm. The survey measurements ranged from 29-63 cpm.

The instrumentation used was listed along with calibration information. The measurement and analysis procedures were described in the report. The survey results were not reported by location but in a summary form. No maps or diagrams were included. The actual survey records were retained for NRC inspection.

Confirmatory Survey Results

A closeout inspection and confirmatory survey was conducted by R.E. Baer, Region IV, on October 11, 1989. The inspector verified the disposition of the fuel and radioactive components. A survey of the reactor components, which were stored in the west pit, indicated exposure rates from 5-20 μR/hr above background on a bolt at the tip of 27 reflector elements. The reflector elements were transferred to the universities byproduct license #35-07466-05. The licenses decommissioning survey log book was found to be in agreement with the final survey report.

Surveys for removable and direct alpha and beta radiation levels were performed. Survey locations included the reactor tank, east storage pit, west storage pit and general area. All direct beta results were less than the detection limit of 0.02 mrad/hr. Direct alpha results were less than the detection limit of 20 dpm/100 cm^2. Removable alpha and beta were less than 20 dpm/100 cm^2 and 400 dpm/100 cm^2 respectively. Confirmatory exposure measurements were reported for each location. No survey exceeded the background rate of 12 μR/hr.

All measurement, sampling, and analysis procedures were described.
The results were reported by location. All instrumentation was listed along with serial numbers.

5. License Termination Date

License R-53 was terminated on February 14, 1990.

6. Federal Register Notice

The termination order was transmitted for publication on February 14, 1990.

7. Hazardous Waste identified at the Site

No hazardous wastes were identified in the docket file.

8. Restrictions/Conditions/Concerns

There are no restrictions indicated in the docket file.

9. Does the Site Meet the Current Criteria?

The removable alpha and beta-gamma contamination criteria in Regulatory Guide 1.86 were satisfied. The exposure rate criteria of 5 μR/hr above background @ 1 meter was also satisfied. Residual direct alpha and beta-gamma radiation levels were not measured by the licensee. The confirmatory survey included a limited number of direct measurements which were all indistinguishable from background.

10. Recommendations

The licensees final survey report did not include direct surface radioactivity measurements. However, the licensees comprehensive smear and exposure rate surveys along with the NRC inspectors direct contamination survey results provide enough evidence to conclude that the site meets the current unrestricted release criteria.

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REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	William Rice University
Address:	Houston, TX
Facility:	AGN-211 Serial Number 101
Site Contact:	R. G. Cochran, Head of Nuclear Engineering
NRC Contact:	D. J. Skovholt, Division of Reactor Licensing
Docket No.:	50-114
License No:	R-54

1. Summary of Operating History

The William Rice University in Houston, Texas obtained a license to operate a AGN-211 reactor on January 8, 1959. The AGN-211 "Portable Pool Reactor" is a swimming pool type reactor containing less than 1 kg of U-235 in the form of 20% enriched UO_2 dispersed in a polyethylene matrix. It had a normal operating power level of 10 watts providing a peak thermal neutron flux of approximately 3 E+08 n/cm^2-sec. The reactor was used for conducting classroom experiments.

In 1965, the university determined that it was not economically feasible to continue operation of the reactor. A new program was set up to utilize the Texas A&M University reactor for the experiments.

The component parts of the AGN-211 reactor were transferred to Texas A&M University to serve as spare parts for their AGN-201 reactor. Fuel elements were sent to the Y-12 Plant in Oak Ridge, TN.

2. Request for Termination of the License

The licensee requested termination of the license on March 10, 1967.

3. Dismantling Order Date

The AEC issued an order authorizing dismantling of the facility on December 2, 1965.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The Docket File contains no information on the primary contaminants.

Applicable Guidelines, Supplemental Limits, Exceptions

The Docket File references 10 CFR 20.

Licensee's Survey

Health physicists from Texas A&M University were present when the reactor parts were packaged. Each item was surveyed prior to packaging and indicated negligible contamination. Debris from the fuel elements was removed from the shield tank and assayed for radioactivity. No detectable activity was indicated; however, the debris was placed in the radioactive waste. Surveys were conducted using a Tracerlab SU-14 alpha-beta-gamma detector.

Health physicists also surveyed the reactor room, adjoining labs, neutron sources, and fission plates for removable contamination. The maximum detectable activity was 6.0 E-05 μCi beta. Smears were also taken from surfaces in the reactor room and labs. The highest levels detected were 6.7 dpm beta, 69 dpm gamma, and 0.18 dpm alpha.

Measurement, Sampling, and Analyses Procedures

The Docket File contains a letter describing the dismantling process and the types of surveys to be performed as dismantling proceeded. Specific procedures on measurement and sampling techniques are not included in the Docket File.

Final Survey Data

Health physicists from Texas A&M University were present when the reactor components were packaged and surveys were performed of each component. Additionally, the reactor room and labs were surveyed (see Licensee's Survey above). The shield water was analyzed and indicated $<$3 E-07 μCi/ml gross beta-gamma and $<$ 2 E-06 μCi/ml tritium.

The site was surveyed by a health physicist from Texas A&M University on August 3, 1967. A radiation survey of Room 132 of Ambercrombie Hall indicated radiation levels were below 2 mrem/h from neutron and gamma radiation. A survey for removable contamination was also conducted in Room 132 and indicated no detectable contamination.

Confirmatory Survey Results

On August 16, 1967 a Radiation Specialist from the Region IV office conducted a survey for fixed contamination in the facility (Rooms 132, 133, 135) and found only background radiation levels ($<$0.2 mrad/h). He used a Tracerlab SU-14 alpha-beta-gamma detector equipped with a TGC-9 mica window ($<$ 2 mg/cm^2) geiger tube. The Safety Evaluation by the Division of Reactor Licensing dated September 26, 1967 stated that an AEC inspector verified that the AGN-211 reactor was dismantled and the component parts and fuel were disposed of. The inspector also verified that the maximum radiation levels at the site were within the 10 CFR 20 limits for unrestricted use and that there was no detectable contamination above background levels.

Future Use of Facility

The Docket file does not indicate the future use of the facility. It would most likely continue to be used for lab and classroom work.

5. License Termination Date

The AEC issued a license termination order on September 29, 1967.

6. Federal Register Notice

A copy of the termination order was forwarded to the Office of the Federal Register for publication on October 6, 1967.

7. Hazardous Waste Identified at the Site

The Docket File does not contain any information on hazardous waste.

8. Restrictions/Conditions/Concerns

The Docket File does not contain any restrictions, conditions or concerns.

9. Does the Site Meet Present Criteria?

There is insufficient radiological survey data in the Docket File to adequately determine that the facility meets the present criteria.

10. Recommendations

The Docket File contains only limited radiological survey data. Survey procedures and maps are not included. If additional survey information is available from Texas A&M University, it should be included in the Docket File. A more thorough survey would be needed to ensure that the present release criteria are met; however, since it has been over 20 years since the license was terminated, it is unlikely that additional surveys would provide meaningful information.

Prepared by: Betty M. Smith
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: June 14, 1991

REVIEW OF TEST AND RESEARCH REACTOR TERMINATED LICENSES

Site:	University of Wyoming
Address:	Laramie, WY 82071
Facility:	Atomics International L-77
Site Contact:	V. A. Ryan, Safety Director
NRC Contact:	R. W. Reid, Division of Operating Reactors
Docket No.:	50-122
License No:	R-55

1. Summary of Operating History

The University of Wyoming obtained a license to operate the Atomics International L-77 solution-type, light water moderated reactor at Laramie, WY on February 25, 1959. The information in the Docket File indicates that all reactor operations were routine and consisted primarily of limited irradiations and maintenance operations. The reactor was operated for a total of 36.25 watt-hours.

The reactor was transferred to the Central Florida Community College at Ocala, Florida. The reactor fuel was transferred to Idaho Falls.

2. Request for Termination of the License

The University of Wyoming applied for termination of license number R-55 on November 1, 1974.

3. Dismantling Order Date

The AEC issued an order to defuel and to possess the reactor on June 11, 1973. The Dismantling Order was not issued at the same time, since disposition of the reactor was not known. The Dismantling Order was issued on July 9, 1973.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The Docket File contains no information on the primary contaminant.

Applicable Guidelines, Supplemental Limits, Exceptions

The Docket File does not reference any specific guidelines or limits.

Licensee's Survey

On August 8, 1974, after the fuel was removed and transferred, the licensee performed surveys for alpha, beta and gamma activity using a Ludlum Model 12 count rate meter. Only background levels were detected. Smears of the reactor floor and reactor surface were also performed and counted using the Ludlum Model 12 with a PPA-2 pancake crystal. Smears indicated background activity.

On October 25, 1974, prior to the transfer of the reactor, a smear survey was performed on the reactor and reactor room floor; the activity ranged from 8 to 12 cpm.

On October 29, 1974, after removal and transfer of the reactor, a smear survey was performed of the reactor room floor; the activity detected ranged from 7 to 12 cpm.

Measurement, Sampling, and Analyses Procedures

The Docket File contains the procedure to be used for dismantling. This procedure indicates that radiological surveys are to be performed, but does not contain any specific survey or sampling procedures.

Final Survey Data

The only surveys mentioned in the report are those performed after fuel transfer, and prior to and after reactor transfer. It is stated in the Docket File that these surveys did not indicate any radiation or radioactive contamination above background levels.

Confirmatory Survey Results

On October 29, 1974, R. F. Warrick of the NRC conducted an inspection of the facility. He verified that the components were packaged and shipped to Central Florida Communit; College. The inspector surveyed the external surfaces of the reactor tank, the crates, the moveable shield, and the floor of the reactor room using a Frieseke Hoepfner portable survey instrument. All direct contact readings were less than 0.1 mR/h. Smears were taken on the external surfaces of the reactor tank, the crates, the shield and the reactor floor area. No detectable contamination was found. The inspector also reviewed the surveys performed by the university.

Future Use of Facility

The Docket File does not indicate the future use of the facility.

5. License Termination Date

The AEC issued the order to terminated the license on December 5, 1974.

6. Federal Register Notice

The Docket File does not contain any information pertaining to the Federal Register Notices.

7. Hazardous Waste Identified at the Site

The Docket File does not indicate whether any hazardous wastes were present.

8. Restrictions/Conditions/Concerns

No restrictions, conditions or concerns are mentioned in the Docket File.

9. Does the Site Meet Present Criteria?

The Docket File indicated that radiation surveys were performed and that only background levels were indicated. The results of the smear surveys were also said to be at background levels and the only actual results presented are give in cpm instead of dpm. No surveys were performed on the walls or ceilings. No samples were taken. Exposure rates r eported are at a detection sensitivity above the guideline of 5 μR/h.

From the information presented in the Docket File, It cannot be determined whether or not the site would meet the present criteria.

10. Recommendations

Obtain records of surveys performed. In order to determine if the facility meets the present requirements, a survey would need to be performed, however, since it has been longer than 20 years since the license was terminated, a survey of the area would probably not provide much information.

Prepared by: Betty M. Smith
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: June 14, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Virginia Polytechnical Institute and State University
Address:	Blacksburg, Va 24061
Facility:	Virginia Tech Argonaut Reactor (VTAR) Facility
Site Contact:	Dr. Keith Furr, Head Health and Safety department
NRC Contact:	George Kuzo, NRC Region II
Docket No.:	50-124
License No:	R-62

1. Summary of Operating History

The Virginia Tech Argonaut Reactor (VTAR) Facility was located in Robeson Hall on the northwest corner of the main campus of Virginia Polytechnic Institute and State University (VPI) in Blacksburg, VA. The reactor was used as part of the Nuclear Engineering curriculum for basic research in neutron physics, neutron radiography, neutron activation analysis, technical training, reactor operator training, and experiments associated with health physics and nuclear engineering.

The VTAR was an Argonaut type research reactor originally designed and installed by American Standard Nuclear Division. The reactor began operation in June 1959. Originally licensed for a maximum power level of 10 kW(th), the reactor was modified and the license amended to allow a maximum power of 100 kW(th) in late 1966. Total accumulated power from 1959-1983 was 1,276 Mw-hrs.

In 1971, an incident occurred during reactor operation which resulted in the release of irradiated U-235 products into areas near Room 10 and Room 106. VPI performed decontamination of these areas.

The reactor was shutdown on July 14, 1983 and a possession only license issued in April 1985. Dismantlement and decommissioning operations began in September 1985 and were completed in January 1987 by Chem-Nuclear Systems, Inc., Columbia, SC.

The reactor fuel was shipped to DOE Savannah River Site in late 1985 and early 1986.

2. Request for Termination of the License

The licensee requested termination of the license for the VTAR Facility on July 17, 1986.

3. Dismantling Order Date

The Dismantling Order was issued on October 29, 1986.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The primary sources of radiation in the VTAR Facility consisted of radioactive material from the reactor structures and components. The concrete rubble was expected to consist of less than 10 curies of primarily Co-60 and Zn-65.

Applicable Guidelines, Supplemental Limits, Exceptions

Regulatory Guide 1.86 and 5 μR/h above background when measured at 1 meter above the surface.

Licensee's Survey

The licensee performed a final survey of the facility and submitted a report to the NRC in July 1986. The survey data indicated that the facility met the guidelines for unrestricted release.

Measurement, Sampling, and Analyses Procedures

The survey plan and survey report provide detailed procedures for radiological measurements and sampling. Analysis procedures are referred to, but not summarized.

Final Survey Data

VPI submitted their final decommissioning report to the NRC in April 1987. The survey appears thorough and indicated that all measurements were below the guideline values.

Confirmatory Survey Results

NRC Region II performed a closeout inspection and survey of the facility on July 27-29, 1987 (NRC Inspection Report No. 50-124/87-01). Oak Ridge Associated Universities (ORAU) performed the confirmatory survey. The survey included surface alpha, gamma and beta-gamma scans, measurement of direct and removable contamination levels, and the measurement of radionuclide concentrations in soil, concrete rubble, and residue samples. The survey findings indicated that the facility met the NRC guideline for unrestricted release.

Future Use of Facility

The Docket File does not indicate a future use for the facility.

5. License Termination Date

The NRC issued the Termination Order on August 11, 1988.

6. Federal Register Notice

Memos to the Federal Register for publication were issued for the issuance of the license on December 3, 1959 and for license termination on August 11, 1988.

7. Hazardous Waste Identified at the Site

No hazardous wastes were expected to be generated.

8. Restrictions/Conditions/Concerns

Data in the ORAU confirmatory survey indicated that the burial of waste material, with radioactive contamination below established unrestricted release limits from the reactor facility, appeared to have resulted in elevated radionuclide concentrations in soil and in elevated exposure levels at the University landfill.

The licensee rearranged parts of the landfill to eliminate the existence of a 4 pi geometry, which resulted in exaggerated measurements. Another survey was performed by Region II on April 2, 1988 (NRC Inspection Report No. 50-124/88-01). Based on this survey, it was concluded that the residual contamination levels for the materials present in the landfill complied with the requirements of Regulatory Guide 1.86.

9. Does the Site Meet Present Criteria?

Based on the final survey performed by the licensee ano 'he confirmatory survey performed by ORAU, the VTAR Facility does meet present release criteria.

10. Recommendations

None.

Prepared by: Betty M. Smith
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: June 14, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site: West Virginia University
Address: P.O. Box 6057, Morgantown, West Virginia
Facility: AGN-211P Research Reactor
Site Contact: William E. Collins, Professor and Chair Radiological Safety Committee
NRC Contact: Darrel G. Eisenhut, Director, Division of Licensing
Docket No.: 50-129
License No: R-58

1. Summary of Operating History

The AGN-211P Reactor was a small research reactor located in Hodges Hall on the Downtown Campus of the University of West Virginia. The reactor was used primarily for research and teaching purposes. It was designed to operated at a maximum power of 75 watts. The reactor operated for approximately 100 hours each year at 75 watts over a 12 year period. The reactor ceased operations in February 1971.

The reactor core consisted of a matrix array of 12 fuel elements surrounded by 30 graphite reflector elements. The core was situated at the bottom of a ten foot deep water-filled tank. The fuel elements consisted of 20% enriched UO_2 pellets fused in a polyethylene moderator. The total fuel loading was 800 g of U-235.

During the closeout inspection two Notices of Violation were issued by the NRC. They involved not maintaining records of surveys conducted during dismantling and disposal of radioactive material as clean material.

2. Request for Termination of the License

A request for termination of the license was submitted to the NRC on September 27, 1979 and supplemented by a letter dated November 30, 1979.

3. Dismantling Order Date

The NRC approved the Dismantling Order on January 22, 1980.

The reactor fuel elements, all control/safety rods, electromechanical rod drive mechanisms and some peripheral hardware were sent to the University of Oklahoma. The remaining parts of the reactor (control console, super structure, etc.) were disposed of.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The Docket File contains no information on the primary contaminants.

Applicable Guidelines, Supplemental Limits, Exceptions

Regulatory Guide 1.86

Licensee's Survey

Between June 11, 1980 and February 7, 1983, eleven wipe surveys were conducted in the reactor site rooms. The only area found to be contaminated was a laboratory hood where radionuclide work was being conducted. The radiation health technician concluded that the contamination was not due to the reactor.

Prior to shipment or disposal of reactor components, radiation surveys and wipe surveys were performed. Additionally, rooms B-30 and B-31 in Hodges Hall, the site of the reactor facility, were surveyed.

Measurement, Sampling, and Analyses Procedures

The Docket File contains no information on measurement, sampling, or analysis procedures.

Final Survey Data

Wipe surveys were performed in the Hodges Hall Vault (Room 132). Results indicated it was "mildly" contaminated. The contamination consisted of tritium on some shelves, floors and walls. The radiation health technician concluded the contamination was not from the reactor components that were stored in the area, since they were wiped tested before and after they left the vault, but was due to storage of radionuclides used in laboratory work.

During dismantlement, wipe surveys were performed using Q-tips and wiping a 100 cm^2 area. Additionally, sludge, rust and dirt samples were collected. All samples were counted for 100 minutes in a Nuclear Chicago Automatic Gamma Well Counter. Elevated levels were detected in one soil and one concrete rubble sample. The radiation health technician concluded the elevated activity was due to background since all regions of the spectrum indicated elevated counts.

Reactor water samples were counted for 10 minutes in a Nuclear Chicago Automatic Gamma Well Counter, a Beckman LS 9000 Scintillation Counter and in an Eberline Lab Counting System LCS-1. The results indicated background levels of activity.

Confirmatory Survey Results

A closeout inspection was conducted by R.H. Albright, Region II NRC, on June 22-23, 1983. The inspector performed confirmatory surveys for direct radiation and contamination in the source storage vault and in the rooms which housed the reactor and control console.

Direct radiation surveys were performed at 0.5 inches from the floors using an Eberline RM-14 with HP-260 pancake probe. Two spots of contamination were identified in the source storage vault. One spot read 320 cpm above background and was fixed. The second spot was at the usual location for storage of the Ra-Be source A smear of this location indicated beta activity at 970 dpm/100 cm^2 and alpha activity at 7 dpm/100 cm^2. Analysis of the smear on a GeLi counter indicated 1.98 E-06 μCi/100 cm^2 U-235 and 2.90 E-05 μCi/100 cm^2 U-238. The NRC concluded that the alpha activity was below the guideline limits and the beta activity may have been due to other sources stored in the vault not associated with the reactor.

Direct radiation measurements were taken at 1 meter using an Eberline PRM-7 μR/h ratemeter. All areas surveyed were below the guideline levels.

All smears were less than 1000 dpm/100 cm^2.

Future Use of Facility

The facility was to be used as a university research facility.

5. License Termination Date

The NRC issued a letter for termination of the license on September 7, 1984.

6. Federal Register Notice

No Federal Register notices were included in the Docket File.

7. Hazardous Waste Identified at the Site

The Docket Files does not identify any hazardous wastes at the site.

8. Restrictions/Conditions/Concerns

No restrictions, conditions, or concerns are noted in the Docket File.

9. Does the Site Meet Present Criteria?

The data presented by the licensee and NRC inspector in the Docket File indicate that the surveys performed meet present criteria. However, the extent of the surveys seems to have included only the floors, lower walls, and equipment. The surveys did not address the upper walls, ceiling, ducts, piping runs, sinks, etc.

10. Recommendations

If additional data are available, they should be reviewed to determine the thoroughness of the survey. A more detailed survey should be performed to verify that the facility was adequately decontaminated.

Prepared by: Betty M. Smith
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: January 30, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site: Walter Reed Arm, Medical Facility
Address: Washington, D.C. 20012
Facility: Research Reactor
Site Contact: Billy G. Bass, Director, Walter Reed Research Reactor
NRC Contact: None Indicated
Docket No.: 50-135
License No: R-85

1. Summary of Operating History

Walter Reed Medical Facility (WR) operated a pool-type research reactor from 1962 to January 1970. The reactor, located in northwest Washington, D.C., was a homogeneous solution type, Atomic International Model L-54 nuclear reactor.

No operating history, including spills, incidents, accidents, or fuel failures was provided in the Docket File.

2. Request for Termination of the License

The Decommissioning Plan required:

- the removal and transport of 26.5 liters of aqueous homogeneous uranyl sulphate fuel solution containing 1.325 kg of U-235 to the AEC's Chemical Processing Plant near Idaho Falls, Idaho,
- removal and transport of radioactive material (e.g., reactor vessel, control rod drive mechanisms, etc.) to a land burial site,
- removal and disposal of the concrete biological shield, reactor controls, and monitoring instrumentation, and
- decontamination of the reactor room to make it suitable for occupancy as an unrestricted access laboratory.

3. Dismantling Order Date

The Dismantling Order was approved by the AEC on July 6, 1971.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The Docket File does not contain any information on the primary contaminants.

Applicable Guidelines, Supplemental Limits, Exceptions

The Dismantling Plan refers to the "AEC Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use, April 22, 1970."

Licensee's Survey

WR conducted a final survey which included exposure rate measurements on all floor and wall areas, with readings taken within 2 inches of the surface. Wipe surveys were performed, but no locations are given.

Measurement, Sampling, and Analyses Procedures

The Docket File does not discuss specific procedures. However, in the close-out inspection report, 50-135/72-01, the Region I inspector (W.H. Baunack) indicated he reviewed documents and procedures concerning administrative controls, radiation monitoring, fuel transfer and shipping, reactor system dismantling and shipment, and facility decontamination.

Final Survey Data

The final survey was conducted by WR personnel on February 10, 1972. The maximum observed exposure rate was 0.5 mR/h, using a Victoreen MD1 440 ion chamber survey meter. Wipes were taken on all floor and wall surfaces. The maximum removable activity indicated was 269 dpm/100 cm^2 in the exposure room. All other wipes indicated $\leq$ 215 dpm/100 cm^2. Wipes were counted on a R/A Lab, NL & TS Br Beckman Wide Beta II. The absolute filter in the lower level read 1 mR/h.

The final survey report recommended that the inner sub-pile room be decontaminated again and the surface be painted with epoxy paint to fix the contamination; additionally, the absolute filters be changed and disposed of since they contained fission and activation products released during dismantling. The final survey report indicated that these results did meet AEC requirements.

The Docket File does not contain any information as to whether the recommendations were carried out.

Confirmatory Survey Results

On March 16-17, 1972, an AEC Region I inspector performed a close-out inspection. The inspection consisted of a review of records, documentation of dismantling and decontamination, and procedures.

No confirmatory survey is indicated in the Docket File.

Future Use of Facility

Unrestricted access laboratory.

5. License Termination Date

The AEC approved termination of the license on July 26, 1972.

6. Federal Register Notice

No Federal Register notices were included in the Docket File.

7. Hazardous Waste Identified at the Site

No hazardous wastes were indicated in the Docket File.

8. Restrictions/Conditions/Concerns

No restrictions, conditions, or concerns were indicated in the Docket File.

9. Does the Site Meet Present Criteria?

The Final survey indicated exposure rates were a maximum of 0.5 mR/h. It does not meet the current 5 μR/hr above background criteria.

The wipe surveys indicated a maximum of 269 dpm/100 cm^2. This would meet present criteria if the primary contaminant was uranium.

The final survey indicated a recommendation to paint over contamination, to reduce activity levels. This action would not be appropriate for meeting present criteria. However, there is no indication whether the recommendations were implemented.

The Docket File does not adequately identify the extent and location of the surveys performed, does not identify if the final survey recommendations were implemented, and does not indicate that a confirmatory survey was performed. Additionally, no mention is made of surveying upper walls or ceilings. Since the absolute filters were contaminated during dismantling and decontamination, it would indicate that contamination may have been airborne and been spread to upper areas.

10. Recommendations

More detailed information on implementation of recommendations and actions would need to be provided and a more detailed survey may need to be performed to verify the adequacy of the decontamination of the site. However, due to the time elapsed since termination of operations and other uses of radioactive materials, if used, that have occurred since license termination, it would be difficult to associate the presence of residual contamination with the R-85 license.

Prepared by: Betty M. Smith
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: January 29, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Leland Stanford Junior University
Address:	University Campus near Palo Alto, California
Facility:	Stanford Pool Reactor
Site Contact:	R. A. Finston, Director, Health Physics
NRC Contact:	D. J. Skovholt, Assistant Director for Operating Reactors
Docket No.:	50-141
License No:	R-60

1. Summary of Operating History

Stanford University was licensed by the AEC on December 4, 1959, to operate a 10 kilowatt, pool-type heterogenous, light water-moderated and reflected, training and research reactor to conduct educational activities. The license approved possession and use of up to 4 kilograms of uranium-235 in the form of fuel elements, 2 grams of uranium-235 as a fission counter, and 80.2 grams of plutonium as a Pu-Be neutron source. The authorized power level was 10 kilowatts.

2. Request for Termination of the License

A request for license termination was filed with the NRC on August 9, 1976. The request included a summary of the results from a licensee survey of the facility.

3. Dismantling Order Date

Authorization for dismantlement of the reactor was requested on November 19, 1973. The AEC filed a notice of intent to issue authorization to dismantle the facility on February 25, 1974. The notice required that all planned liquid releases, be verified within 30 days, and be within the standards established by 10 CFR Part 20. A dismantling order was issued by the AEC on March 12, 1974.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The licensee was authorized to possess and use up to 4.0 kilograms of contained uranium-235, 2 grams of uranium-235 as a fission counter and 80.2 grams of plutonium as a Pu-Be neutron source in connection with operation of the reactor. Co-60, Eu-152, and Cs-137 were mentioned by the NRC as isotopes of concern.

Applicable Guidelines, Supplemental Limits, Exceptions

The NRC determined that surfaces of the facility must be decontaminated to satisfy Reg. Guide 1.86 and that Co-60, Eu-152, Cs-137, or any reactor generated gamma emitting isotopes must be removed such that radiation levels are less than 5 μR/hr above natural background measured at one meter from the surface.

The safety evaluation conducted in support of the AEC's notice of intent to issue authorization for dismantling determined that the release of pool water or other liquids be based on limits imposed by 10 CFR Part 20. A letter from the NRC on April 21, 1982, stated that radiation from gamma emitting isotopes is also acceptable if the potential exposure to individuals is less than 10 mrem per year with reasonable occupancy assumptions. In reference to concrete samples, 10 CFR 30.70 Schedule A was noted as the applicable criteria.

Licensee's Survey

The Docket File indicates that the biological shield and the immediate proximity around the reactor was to be surveyed, to include the collection of swipes after fuel removal and, thereafter, at quarterly intervals by the Health Physics Department; however, results have not been included in the file. Analytical results for one liquid sample of unspecified origin contained an alpha concentration of 0.27×10^{-8} μCi/ml.

In August and September of 1973, the University Health Physics group conducted surveys to assess radiation and contamination levels. Storage area radiation levels were <5 mrem/h at 1 foot from sealed sources, and less than 0.25 mrem/h in occupiable areas. Removable activity levels around the biological shield were <100 dpm/100 cm^2 alpha and <1000 dpm/100 cm^2 gross beta. Radiation levels in the same area were reported to be <0.2 mrad/h at 1 cm. The activity level in the pool was measured in May of 1973 and found to be <100 mrad/h. The gross alpha activity in the reactor pool was also measured to be less than the activity for discharging into uncontrolled areas as described in 10 CFR 20.

The file indicates that the licensee performed a survey from September 1975 to May 1976 for fixed and removable contamination. The results showed all areas to be <0.03 mrem/hr at 1 cm and smear activity showed <100 dpm/100 cm^2. Concrete samples taken from the floor of the reactor tank structure ranged in activity from 0.36 to 45.80 pCi/g for Co-60 and 0.48 to 81.23 pCi/g for Eu-152.

Measurement, Sampling, and Analyses Procedures

The Docket File contains some description of the gamma spectrometry procedure used to analyze concrete samples. Some licensee survey forms include a description of procedures for removable contamination sampling and exposure rate measurements. A general procedure by Radiation Detection Company for the analysis of a liquid sample was also included in the file.

Final Survey Data

Concrete samples were collected on October 13, 1976, from the floor under the reactor tank over which the reactor core was suspended. The most active sample contained 1.168×10^{-4} μCi/g $\pm$ 2.48% of Co-60 and 1.653×10^{-4} μCi/g $\pm$ 5.12% of Eu-152. Direct measurements of surface activity were made on December 3, 1976 and March 21, 1977. The maximum reading was found to be 420,000 dpm/100 cm^2; the average reading was 109,200 dpm/100 cm^2 measured with a GM meter.

No detectable removable contamination was identified. Measurements made on December 8, 1977, using a Health Physics Instruments, Inc. Model 1010, tissue equivalent ion chamber, determined that the maximum level was 40 μrads/h at 1 meter from the floor of the reactor.

A letter from the licensee dated June 1, 1988, informed the NRC of plans to demolish the building which housed this reactor. Additional core samples were removed and analyzed for Eu-152 and Co-60. Results from two samples were included in the letter. Sample #01 contained 150 pCi/g Eu-152 and 41.3 pCi/g Co-60. Sample #02 contained 51 pCi/g Eu-152 and 12.5 pCi/g Co-60. The licensee indicated that they planned to remove a two inch layer of grout from the area of concern and dispose of it as radioactive waste. The remainder of the biological shield, the floor of the laboratory, and the adjacent concrete office building, were to be demolished and crushed and the crushed rock released for unrestricted use. At the conclusion of this demolition, soil sampling was to be done to verify that no residual activity remained.

Confirmatory Survey Results

On January 31, and March 13, 1979, the NRC conducted an inspection of the facility. The inspector noted that the facility had been deactivated and the reactor core, controls, components, and support laboratory facilities had been dismantled and removed from the facility. It was noted that surveys performed in 1975 showed that the only remaining radioactivity above background levels was located inside the biological shield in the concrete floor that was beneath the reactor core. Observations were made of further concrete core sampling and analysis techniques.

An inspection on March 10, 1981, included discussion about the possibility of allowing remaining activity to decay in place to acceptable levels. It was suggested that the licensee submit a report to the AEC on the potential exposure level.

An inspection on September 16, and November 5, 1982, included the collection and analysis of 67 smears for gross alpha and beta activity. Results showed <6 dpm/100cm^2 alpha and <28 dpm/100cm^2 beta which were noted to be statistically near background. Gamma surveys identified areas above background inside the biological shield only. The maximum dose rate measured by the NRC was 20 μR/h above background.

An on-site gamma spectrum was gathered with a portable intrinsic germanium detector and a Nuclear Data ND-6 portable multichannel analyzer. Eu-152 and 154, and Co-60 peaks were identified. A Dose Equivalent-Rate and commensurate annual exposure calculation was issued by the NRC supporting the facility terminating order. The calculated dose to an individual occupying the area for 20 hours per year was determined to be 0.4 mrem/yr.

Future Use of Facility

The file indicates that the building is likely to be used for engineering and scientific research projects in the future.

5. License Termination Date

The NRC issued an Order Terminating Facility License on June 21, 1983.

6. Federal Register Notice

Notice of Issuance of Facility License was published in the Federal Register on November 5, 1969. A Notice of Intent to Issue Order Authorizing Dismantling of Facility was sent to the Federal Register on January 31, 1974.

An Order Authorizing Dismantling of Facility was sent to the Federal Register on March 13, 1974.

A Notice of Proposed Issuance of Order Authorizing Termination of Facility License was sent to the Federal Register on January 23, 1978. No Federal Register Notice for facility license termination is included in the Docket File.

7. Hazardous Waste Identified at the Site

No hazardous wastes were indicated in the Docket File.

8. Restrictions/Conditions/Concerns

None were noted in the Docket File.

9. Does the Site Meet Present Criteria?

The removable contamination levels reported satisfy the present guidelines. Total activity levels measured in the area of the concrete pad over which the reactor core was suspended did not meet the guidelines. Exposure rate values measured in this area did not meet the guidelines.

10. Recommendations

Confirmation of the demolition activities scheduled by the licensee, to occur in 1988, should be obtained. If those activities have been completed final exposure rate measurements should be obtained from the former reactor core area. Since the concrete beneath the grouted surface was to be disposed of as regular waste, activity levels on this surface should be documented.

Prepared by: A. T. Payne
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: February 7, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Rockwell International Corporation
Address:	Canoga Park, California
Facility:	Fast Critical Experimental Laboratory Reactor
Site Contact:	H. Weiseneak, Director, Research and Engineering
NRC Contact:	H. E. Book, Chief Fuel Facility and Materials Safety Branch
Docket No.:	50-147
License No:	CX-17

1. Summary of Operating History

From October 3, 1960 to September 22, 1973, Rockwell International was licensed to operate a critical facility at their Nuclear Development Field Laboratory in Canoga Park, CA. The original operating license was issued to North American Aviation Incorporated. The reactor was a split-table assembly containing a matrix of 2 inch square aluminum drawers to hold the fuel and moderator material. Various fuel configurations of U-235, U-233, U-238, and thorium were used. The power level was limited to 200 watts. Most of the source and special nuclear materials produced were shipped to Argonne National Laboratory. Recoverable materials were shipped to recovery sites; if not, it was buried as radioactive waste. Small amounts of privately owned materials were held under license No. SNM-21 at the Rockwell headquarters facility storage vault. No operating history of any spills, incidents, accidents, or fuel failures were identified in the file.

2. Request for Termination of the License

The August 30, 1973 application for authority to surrender voluntarily the facility license CX-17 states, that possession of limited quantities of radioactivity from activated materials will be transferred to California Radioactive material license 0015-59 which identifies the radioactive materials used by the facility. A request to possess but not operate the facility was issued September 22, 1973. Rockwell International submitted a dismantling plan to the AEC on July 16, 1974. The plan called for disassembly and decontamination of the critical machine structure. Components and materials which could not be decontaminated were to be disposed of at a licensed radioactive material disposal site or used in projects where limited radioactive material is acceptable with appropriate authorization. Components meeting the requirements for unrestricted use were to be released to Atomics International's Property Management Department.

3. Dismantling Order Date

The AEC issued the Dismantling Order on November 1, 1974.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The Docket File contains California Radioactive Material License No. 0015-59.

Applicable Guidelines, Supplemental Limits, Exceptions

The Docket File indicated that the applicable guideline was AEC Regulatory "Guidelines for Decontamination of Facilities and Equipment Prior to Release for

Unrestricted Use or Termination of Licenses for By-product, Source or Special Nuclear Material," April 22, 1970.

Licensee's Survey

The application for authority to surrender the facility license dated August 30, 1973, includes a support statement summarizing the status of the facility. All source and fuel materials had been transferred to the "SS Storage Vault" at the Nuclear Development Fuel Laboratory. The critical machine was stored in the FLEL. Maximum contamination levels were stated to be <50 dpm/100 cm^2. Removable alpha contamination levels on fuel drawers ranged up to approximately 500 dpm/100 cm^2 and averaged about 200 dpm/100 cm^2. Slight activation of fuel drawers and the critical assembly structure with Co^{60}, Zn^{65}, Fe^{66}, and Mn^{54} were noted. Maximum radiation levels at inner machine faces were up to 0.8 mR/hr and averaged about 0.5 mR/hr. All contaminated materials, with the exception of the critical machine were cleaned or removed as radioactive waste. A statement indicates that the entire facility was surveyed for fixed and removable contamination. Results of the survey indicated that removable levels were <50 dpm/100 cm^2 alpha and <100 dpm/100 cm^2 beta and that total activity levels were 500 dpm/100 cm^2 alpha and 0.1 mrad/hr beta. It is stated that the building meets the AEC "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-product, Source or Special Nuclear Material." Radioactive liquid was removed from waste tanks and disposed of as radioactive liquid waste.

The interior tank surfaces were surveyed and found to comply with the guidelines as stated above. Prefilters and HEPA filters were disposed of as radioactive waste. The filter plena were decontaminated to a level below the above stated guidelines.

The Docket File does not contain any information indicating measurement and sampling locations.

Measurement, Sampling, and Analyses Procedures

The Docket File does not contain any information on measurement, sampling and analysis procedures.

Final Survey Data

The licensee states that a survey was conducted using a Ludlum model 125 Micro R meter and a Technical Associates PUG-1 countrate meter with a thin window pancake GM detector for general background measurements. Residual activity in the critical assembly area was measured with a Technical Associates Model CP-7M ion chamber dosimeter. A Ludlum Model 12 countrate meter with an alpha scintillation probe model 43-5 was used for surveys of alpha activity. Residual activity in the critical assembly areas was <0.1 mrad/hr at 1 cm from the surface with a 7 mg/cm^2 absorber. Smear results were reported by room for alpha and beta-gamma. Smear results in all areas showed <20 dpm/100 cm^2 alpha and <50 dpm/100 cm^2 beta. Analysis for liquid materials revealed 1.9 X 10^{-8} μCi/ml or less alpha and 1.7 X 10^{-8} μCi/ml beta. The Docket File does not contain information concerning survey dates or measurement and sampling locations.

Confirmatory Survey Results

The NRC Office of Inspection and Enforcement conducted an inspection of the facility on June 4, 1980. The inspector conducted independent monitoring and surveys at the facility to verify the adequacy of decontamination activities. The inspection included an examination of records, personnel interviews, and performance of direct radiation and contact measurements. The report states that radiation levels measured within three inches of surfaces were 5 to 50 μR/hr and levels at three feet from surfaces measured <5 to 15 μR/hr using E520 and PRM-7 portable survey instruments. Surface contamination measurements made using a thin window "pancake" GM detector were found to be 10 to 20 cpm. No calibration information is included to allow for conversion of data to units that are applicable to guidelines, however, a statement indicated that no levels >2 times the background range of 10 to 20 cpm were observed. Swipes were taken at 17 locations which were identified on a facility map. No removable surface contamination was detected.

A soil sample from a natural drainage pathway northwest of the facility, and a water sample from pipes in the "hot" drain system were collected. No significant radioactivity was detected by gamma spectroscopy analysis.

Future Use of Facility

The proposed future use of the facility is of a non-nuclear character.

5. License Termination Date

The AEC issued a letter terminating the license on October 1, 1980.

6. Federal Register Notice

The Federal Register published a "Notice of Proposed Issuance of Order Terminating Facility License" on July 25, 1980.

7. Hazardous Waste Identified at the Site

No hazardous waste was indicated in the Docket File.

8. Restrictions/Conditions/Concerns

No restrictions, conditions, or concerns were noted in the Docket File.

9. Does the Site Meet Present Criteria?

The removable contamination levels reported by the licensee were verified by the NRC inspection and satisfy the present criteria. Information concerning total activity levels was not provided in guideline units of dpm/100 cm^2 and no instrument calibration information was provided to allow for conversion to those units.

The exposure rate data provided by the NRC inspection is sufficient to satisfy the present criteria for reactor facility decommissioning of 5 μR/h above background.

The Docket File does not adequately identify the locations of surveys performed. No procedures are included.

10. Recommendations

An attempt could be made to obtain calibration information to convert NRC total activity measurements to guideline units to determine compliance with present criteria.

Prepared by: A. T. Payne
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: February 5, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Westinghouse Reactor Evaluation Center
Address:	Waltz Mill, PA
Facility:	LRX Critical experiments facility
Site Contact:	C.H. Weaver, Vice President, Westinghouse Atomic Power Division
NRC Contact:	E.R. Price, AEC Division of Licensing and Regulation
Docket No.:	50-153
License No.:	CX-16

1. Summary of Operating History

The LRX facility was licensed on May 3, 1960, to operate at up to 3000 watts (thermal) as a utilization facility and for conduct of critical experiments. The license and other documents in the Docket File do not contain any information regarding the design of the facility, the fuel utilized, or other radioactive materials used at the facility. No information concerning spills, incidents, fuel failures, etc. is provided. In a letter of April 8, 1963, requesting license termination, it is stated that the facility has already been disassembled and all source, special nuclear, and byproduct material disposed of. It is also noted that an AEC contract activity incorporating another critical facility, will be conducted at the site.

2. Request for Termination of the License

A request for license termination was submitted by Westinghouse on April 8, 1963. No decommissioning plan was submitted; the facility had already been dismantled and licensed materials disposed of.

3. Dismantling Order Date

No dismantling order was issued.

4. Radiological Status Information

No information regarding the radiological status of the facility is provided in the Docket File. The only related information is the statement that all licensed materials have been disposed of. A confirmatory survey or inspection, supporting the termination action was apparently not performed.

5. License Termination Date

The AEC issued a notice of license termination on April 24, 1963.

6. Federal Register Notice

A notice of license termination was submitted to the Federal Register on April 24, 1963.

7. Hazardous Wastes Identified at the Site

No hazardous waste was indicated in the Docket File.

8. Restriction/Conditions/Concerns

None were identified in the Docket File.

9. Does the Site Meet Present Criteria?

No data regarding the radiological status of the facility are provided in the Docket File; an assessment of the status relative to present criteria is, therefore, not possible.

On April 24, 1963 Westinghouse and the AEC entered into a contract for further activities at this site.

10. Recommendations

Although there is insufficient information in the Docket File to confirm the radiological status relative to the present guidelines, another critical facility was operated at the site under AEC contract, following termination of the CX-16 license. It is unlikely any residual contamination, if present, could be identified as associated with the CX-16 license activities. Documentation on the closeout of the AEC contract operation can serve to confirm the status of this facility.

Prepared by: J.D. Berger
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: December 26, 1990

REVIEW OF TEST AND RESEARCH REACTOR TERMINATED LICENSES

Site:	Martin Marietta Corporation
Address:	Middle River, Maryland
Facility:	Liquid Fluidized Bed Reactor Critical Experiments Facility
Site Contact:	Richard H. Boutelle, Health Physics Section
NRC Contact:	Saul Levine, Division of Reactor Licensing
Docket No.:	50-154
License No:	CX-18

1. Summary of Operating History

On December 28, 1961, Martin Marietta Corporation was issued authorization to operate the Liquid Fluidized Bed Reactor (LFBR) on their corporations site near Middle River, Maryland. The reactor did not achieve criticality and operations were terminated. The fuel pellets leading was removed and stored in a vault at the site.

On December 11, 1962, Martin Marietta applied for authorization to possess and make changes and alterations to the existing facility, but not to operate the facility. The AEC requested more details as to proposed changes before authorization could be given. The facility license was amended on March 13, 1963, to allow possession but not operation of the reactor for five years and possession without loading of the special nuclear fuel material required for eventual operation of the reactor.

Amendment 2 to the license dated November 6, 1963, allowed for possession but not operation of the facility and possession of the residual special nuclear material contained in the experimental loop. This amendment also prohibited the removal of residual fuel in the loop without written authorization by the AEC. In response to the licensee's request for license termination the AEC requested an estimate of the total amount of uranium and uranium-235 still in the loop system and drain tank as well as information about safeguards that were being taken to prevent unauthorized personnel from entering the loop area. Both concerns were addressed in a December 17, 1963, letter to the AEC. Authority to remove the residual entrained fuel was given on February 12, 1964.

2. Request for Termination of the License

A request for termination of this license was made pending dismantling and disposition of the reactor loop and associated equipment.

An outline of the LFBR removal project was included in the Docket File.

3. Dismantling Order Date

Dismantling authorization was given to Ebasco Services, Inc. on July 14, 1965.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

No information identifying primary contaminants was provided in the Docket File.

Applicable Guidelines, Supplemental Limits, Exceptions

The Docket File does not identify the applicable guidelines.

Licensee's Survey

The Docket File does not contain any information concerning licensee surveys.

Measurement, Sampling, and Analyses Procedures

The Docket Files does not contain any information on measurement, sampling and analyses procedures.

Final Survey Data

A letter dated December 1, 1964, from Martin Marietta to the AEC includes the results of surveys of the LFBR abrasion and critical experiment loops. The abrasion loop was determined to be uncontaminated. The critical experiment loop was determined to be contaminated with UO_2. No specific data were provided in the Docket File. Some instrument readings in counts per minute and some smear results in disintegrations per minute were listed but no calibration information was provided.

Confirmatory Survey Results

AEC inspectors verified that the LFBR had been dismantled and, with the exception of the storage and drain tank, the components transferred to off-site storage or disposal areas. The inspectors also verified that maximum radiation levels were within 10 CFR Part 20 limits for unrestricted areas and that there were no detectable contamination levels above normal background in the test and control room. No confirmatory radiological data are included in the Docket File.

5. License Termination Date

The AEC approved termination of the license in February 7, 1966.

6. Federal Register Notice

The notice of license termination was submitted to the Office of the Federal Register for publication on February 7, 1966.

7. Hazardous Waste Identified at the Site

No hazardous waste was indicated in the Docket File.

8. Restrictions/Conditions/Concerns

The notice of termination of facility license indicates that the LFBR was dismantled except for a contaminated storage and drain tank which was to be retained and incorporated under Facility License No. CX-7, Docket No. 50-38.

9. Does the Site Meet Present Criteria?

No data concerning total or removable activity levels or exposure rates are provided in the Docket File, therefore, assessments relative to guidelines cannot be made.

10. Recommendations

A review of available survey data may provide adequate information to make a reasonable assessment of the facility. However, due to the length of time that has passed since the dismantling, it is unlikely that this information is available.

A survey would be required to verify that exposure rates are less than 5 μR/h above background and that residual total surface activity satisfies guidelines. Information obtained from such a survey may not be meaningful as a result of other activities conducted in this facility that may have involved radioactive materials.

Prepared by: Betty M. Smith
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: January 29, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site: Lockheed Aircraft Corporation
Address: Dawsonville, GA - Air Force Plant 67
Facility: LNP Training Reactor
Site Contact: W.A. Pulver, Vice Pres. and General Mgr. Lockheed
NRC Contact: R.L. Kirk, AEC Division of Licensing and Regulation
Docket No.: 50-167
License No.: R-68

1. Summary of Operating History

An operating license was issued on July 22, 1960, authorizing possession of up to 2.905 kilograms of U-235 as elements and a PuBe neutron source containing 80 grams of plutonium and operation to a power level of 10 watts.

Very little additional information is provided in the File. Based on the Docket File it appears that operations ceased on August 15,1960, and the reactor was sent to Argentina for use in the AEC South American Exhibit.

A document dated April 13, 1962, notes return of 52.23 grams of U-235 in fuel plates from the South American Exhibit. It also notes that the license expired September 1, 1960, and shipment of fuel plates should be covered by valid license SNM-260.

2. Request for Termination of the License

No request included in the Docket File.

3. Dismantling Order Date

None provided in the Docket File.

4. Radiological Status Information

No radiological survey information is included in the file.

5. License Termination Date

The license terminated on September 1, 1960, as a result of no further actions. There is no official notice of termination in the Docket File.

Possession of fuel transferred to license SNM-260.

The ultimate disposition of the reactor and its fuel are not described in the Docket File.

6. Federal Register Notice

No notice is indicated in the Docket File.

7. Hazardous Wastes Identified at the Site

No hazardous wastes were indicated in the Docket File.

8. Restrictions/Conditions/Concerns

None indicated in the Docket File.

9. Dose the Site Meet Present Criteria?

Information in the Docket File is insufficient to determine radiological status of the building housing the facility.

10. Recommendations

Pursue followup of closeout of license SNM-260 for information on facility radiological status.

Obtain additional description of the history of reactor use and current status or eventual disposition.

Prepared by: J.D. Berger
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: December 26. 1990

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Lockheed Aircraft Corporation
Address:	Air Force Plant No. 67, Dawson County, Georgia
Facility:	Radiation Effects Reactor
Site Contact:	M. A. Dewar, Scientist
NRC Contact:	D. J. Skovholt, Division of Reactor Licensing
Docket No.:	50-172
License No:	R-86

1. Summary of Operating History

On July 20, 1962, Lockheed Aircraft Corporation was licensed to operate a heterogenous pressurized water-type nuclear reactor. A description of the facility design is included in the Technical Specifications. The license authorized receipt, possession, and use of up to 18.8 kilograms of contained uranium-235 in connection with reactor operations. Three scrams occurred during the period of August 1, 1969 through July 31, 1970. The cause was listed as a malfunction of #3 Safety Channel Converter resulting in a shift of the scram set point. In addition, the reactor was shut down for six days to investigate and correct a control rod that was stuck in the extreme lowered position.

2. Request for Termination of the License

A request for license termination was submitted on June 17, 1971.

3. Dismantling Order Date

Authority to decommission the reactor was requested on April 12, 1971. The request included a decommissioning plan to greatly reduce surveillance requirements and allow part or all of the site to be released for unrestricted occupancy. The AEC authorized dismantling of the facility on June 7, 1971.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The Docket File does not contain any information identifying the primary contaminants.

Applicable Guidelines, Supplemental Limits _ ceptions

A letter to the AEC on February 2, 1971, requests that, the "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-product, Source or Special Nuclear Material," April 22, 1970, be approved as an acceptable standard.

Licensee's Survey

During the period of August 1, 1969 through July 31, 1970, soil samples collected around the reactor facility, contained activities ranging from 4.1×10^{-6} to 28.6×10^{-6} μC/g. Airborne activity was reported to be 0.64×10^{-12} to 3.35×10^{-12} μCi/cc around the facility for the same time period. Control measurements showed 0.35×10^{-12} to 2.02×10^{-12} μCi/cc. Continuous water sampling of the Etowah River was also performed during this period. The concentrations ranged from 1.9×10^{-8} to 6.0×10^{-8} μCi/ml downstream.

Individual locations of measurements, procedures, or specific data are not provided in the file.

Measurement, Sampling, and Analyses Procedures

The Docket File does not contain any information on measurement, sampling, and analysis procedures.

Final Survey Data

No post decontamination survey data is included in the Docket File.

Confirmatory Survey Results

The AEC conducted an inspection of the facility on July 14, 1971. The inspection included examination of documents, procedures and records, interviews with facility personnel, and observations by the inspector.

The inspector noted that irradiated fuel and control assemblies were shipped to Savannah River Project, control rod drives were disposed of as radioactive waste by Chem-Nuclear Corporation, and irradiated fuel assemblies were shipped to Oak Ridge. Two Lockheed Tour Reactors were purchased by the AEC Division of Technical Information and both units were relocated in Oak Ridge.

No items of noncompliance were detected. It was noted that radioactive material remaining as contaminants would be covered under Licence no. GA-95-1.

No confirmatory survey data is included in the Docket File.

Future Use of Facility

5. License Termination Date

   The AEC issued a License Termination Order on August 31, 1971.

6. Federal Register Notice

   Federal Register Notice of the AEC Order Authorizing Partial Dismantling of the Facility was published on June 9, 1971. No License Termination Order publication was indicated in the Docket File.

7. Hazardous Waste Identified at the Site

   No hazardous waste was indicated in the Docket File.

8. Restrictions/Conditions/Concerns

   Radioactive material license No. GA-95-1 will cover the remaining radioactive contaminants at this facility.

9. Does the Site Meet Present Criteria?

No post decontamination survey data is provided in the Docket File, therefore, no assessments of parameters relative to guidelines can be made.

10. Recommendations

None. Residual activity transferred to State of Georgia, License GA-95-1.

Prepared by: A. T. Payne
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: February 6, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	National Aeronautics and Space Administration (NASA)
Address:	Cleveland, Ohio
Facility:	Zero Power Reactor Facility
Site Contact:	B. T. Lundin, Director, NASA
NRC Contact:	D. S. Skovhott, AEC Division of Reactor Licensing
Docket No.:	50-197
License No:	CX-21

1. Summary of Operating History

On February 4, 1964, the AEC issued a license to NASA to operate the Zero Power Reactor II located in the Zero Power Reactor (ZPR) facility at the Lewis Research Center in Cleveland, Ohio. The ZPR facility consisted of 4 rooms, the control room, the reactor room, the solution room, and the locker room. NASA was licensed to possess up to 53 K grams of uranium 235 in connection with operation of the ZPR II reactor.

The ZPR II reactor shared the same facility with the ZPR I reactor. The same control console was used for operating both reactors, therefore, only one reactor could operate any time. The reactors were used primarily for experimental reactor physics research. The ZPR II operated until August 1, 1972.

2. Request for Termination of the License

NASA submitted a request for termination February 1, 1973. The request included a plan for dismantling the reactor facility.

3. Dismantling Order Date

The AEC issued to NASA an order to dismantle the Zero Power Reactor Facility on March 30, 1973.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

NASA performed contamination surveys of the facility during the dismantling process, however, the primary contaminant was not identified.

Applicable Guidelines, Supplemental Limits, Exceptions

Residual activity levels at the ZPR were compared with the AEC Guidelines for facilities and equipment issued April 22, 1970.

Licensee's Survey

The licensee performed surveys and continuous monitoring during the dismantling period in 27 locations in the reactor room and 11 locations in the solution room were surveyed weekly during dismantling operations until all fuel had been removed. The control room, hallway and adjacent annex were surveyed weekly in 13 locations. No smears collected in these areas exceeded the limits of 10 dpm/100 cm^2 alpha or 400 dpm/100 cm^2 beta-gamma during this survey period.

Measurement, Sampling, and Analyses Procedures

A brief description of the decontamination procedures and survey procedures were provided in the Final Radiation Survey Report prepared by NASA.

All items were removed from the ZPR facility prior to decontamination. All surfaces and remaining structures in the reactor room were sandblasted and the steel floors wet mopped. Surfaces were scrubbed in the solution room.

Smears were collected and counted using a thin window ($<150\ \mu g/cm^2$) proportional counter. Total contamination measurements were obtained using a portable alpha scintillation counter (window thickness 1.5 $\mu g/cm^2$). The alpha scintillation detectors were calibrated using a Pu-239 source.

Final Survey Data

The licensee performed a final survey of the facility. The information provided includes figures of the floors and walls of each room sample and measurement location, and survey data. Actual data were only provided for the solution room hold-up tank, piping lights and duct on this area, and for several miscellaneous items in the reactor room.

Total activity levels ranges from 0 to 5600 dpm/100 cm^2 and removable activity ranged from 0 to 400 dpm/100 cm^2. A lower limit of detection was not provided in the data.

Confirmatory Survey Results

The Docket File did not indicate that a confirmatory survey had been performed.

Future Use of Facility

The future use of the facility was not developed in the contents of the Docket File.

5. License Termination Date

The AEC terminated license CX-13 November 13, 1973.

6. Federal Register Notice

Notice of publication in the Federal Register was not provided in the Docket File.

7. Hazardous Waste Identified at the Site

Hazardous Wastes or use of hazardous materials were not discussed in the Docket File. It is difficult to discern if such a problem existed.

8. Restrictions/Conditions/Concerns

None

9. Does the Site Meet Present Criteria?

Residual activity levels reported in the final survey would be in accordance with present release criteria. However, since data were not reported in the tables provided for all the areas surveyed, it is difficult to confirm that the control room, reactor room, solution room, locker room and any access hallways meet the present criteria.

10. Recommendations

It may be difficult to obtain the original survey data for review. A survey of the former ZPR facility by the current occupant, NRC Region III, or agent of the NRC, would determine whether the radiological status satisfies the current guidelines.

Prepared by: P. R. Cotten
Oak Ridge Associated Universities
Oak Ridge, TN 37830
Date: January 29, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	University of Nevada
Address:	Reno, Nevada
Facility:	Nuclear Reactor Facility - Atomics International Model L-77 Reactor
Site Contact:	Mr. N. Ed Miller - President, University of Nevada
NRC Contact:	Mr. Donald T. Skovholt - AEC, Director of Reactor Licensing
Docket No.:	50-202
License No:	R-91

1. Summary of Operating History

The University of Nevada in Reno, Nevada, operated a Model L-77 solution type nuclear reactor for the purpose of conducting research studies and instructional purposes. The University was granted a license to operate the reactor on September 20, 1963. The license allowed the University to possess 1350 grams of U-235 and restricted power levels to 10 watts. The reactor was located in the Nuclear Reactor Facility Building, a metal Butler type building with 2560 ft^2 of usable floor space.

The operating history of the reactor was not provided in the documentation. The data of final operation of the reactor was not provided. The Docket File did not contain information of any events of spills or accidents.

2. Request for Termination of the License

An application for termination of license R-91 was submitted to the AEC on May 23, 1973. The application was notarized July 24, 1973, and a supplement to the application was provided September 20, 1973. A plan for the dismantling and decontamination of the reactor and facility was submitted at this time. The application indicated that the reactor and components would be given to the University of California at Santa Barbara, Santa Barbara, CA.

3. Dismantling Order Date

The AEC authorized the University to dismantle the reactor on December 21, 1973.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The University did not perform specific analyses to determine the primary contaminant; only smears for gross alpha and beta were collected.

Applicable Guidelines, Supplemental Limits, Exceptions

The AEC indicated to the University that the "Guide for Decontamination of Facilities...," dated April 22, 1970, should be used in preparation for dismantling and decontamination in a letter dated December 20, 1971.

Licensee's Survey

The Docket File does not contain any information about the radiological status of the Nuclear Reactor Facility prior to decommissioning activities.

Measurement, Sampling, and Analyses Procedures

Information was provided pertaining to sampling and counting procedures in the "Contamination Report" prepared by the University. Detection and counting systems were not identified.

Final Survey Data

The University provided a final radiological status report of the Nuclear Reactor Facility in the "Contamination Report" dated September 9, 1974. The report provided information pertaining to the characteristics of the reactor facility and provided figures to indicate building areas and sampling locations. A total of 301 smears were collected throughout the facility. All but five of the smears had little or no detectable activity. However, significant activity was identified in the Reactor Room Vault. Sources were stored in this room during the period that the reactor was operational. Data provided in the report indicate that removable activity levels on the floor of the Reactor Room Vault for alpha and beta ranged from 417 dpm/100 cm^2 to 91,123 dpm/100 cm^2, and 3717 dpm/100 cm^2 to 812,821 dpm/100 cm^2, respectively. The University proposed to leave the vault area intact and locked until such time that the University's physical plant personnel could jackhammer the area to remove the contamination. The vault was later demolished (correspondence of January 15, 1975.)

Confirmatory Survey Results

The Docket File did not contain any information to indicate that an independent confirmatory survey was performed.

Future Use of Facility

The Nuclear Reactor Facility is to be used by the Department of Agriculture.

5. License Termination Date

The AEC terminated license R-91 on February 24, 1975.

A note to the Docket File states that the license would be terminated following the removal of fuel from the fuel storage vault. The University provided confirmation that the removal of the fuel was complete (January 15, 1975).

6. Federal Register Notice

Not included in the Docket File.

7. Hazardous Waste Identified at the Site

Information on hazardous wastes was not provided in the Docket File.

8. Restrictions/Conditions/Concerns

Nothing was identified in the Docket File.

9. Does the Site Meet Present Criteria?

Exposure rate measurements were not performed as part of the University's final survey. The survey results provided in the contamination report did not contain direct or fixed measurement information. All smears for removable activity, except for the five smears collected from the Reactor Room Vault, meet the guidelines of 1000 dpm/100 cm^2 (removable). THe vault was, however, demolished following this survey.

10. Recommendations

A complete radiological survey of the former Reactor Room Vault area should be performed to assess the current status, and exposure rate measurements should be obtained throughout the facility.

Prepared by: Phyllis Cotten
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: February 7, 1991

REVIEW OF TEST AND RESEARCH REACTOR TERMINATED LICENSES

Site:	General Electric Company
Address:	Building 105, Vallecitos Atomic Laboratory, Alameda County, California
Facility:	Mixed Spectrum Critical Assembly
Site Contact:	B. D. Wilson, Administrator and Licensing
NRC Contact:	P. A. Morris, Director, Division of Reactor Licensing
Docket No.:	50-203
License No:	CX-20

1. Summary of Operating History

The Mixed Spectrum Critical Assembly (MSCA) was constructed by General Electric (GE) for the AEC and was licensed for operation on March 29, 1968. The MSCA was within a critical assembly cell in the Critical Experiment Facility located in building 105 of G.E.'s Vallecitos Atomic Laboratory in Alameda County, California.

The facility is described as a low power mock-up of a nuclear reactor in which boiling and superheating of the resultant steam took place. The approved power level was 2 Kw. It was used for critical experiments pertinent to the Mixed Spectrum Superheat Critical program and pulsed neutron source experiments. The Docket File states that no unsafe or potentially unsafe condition occurred during the 303 hours of use. Some technical specifications and general design considerations are included in the Docket File.

2. Request for Termination of the License

Application for license termination was submitted on January 8, 1968, by General Electric Company. The application included a brief discussion of dismantling and disposal activities.

In response to this request the AEC informed General Electric Company that prior to termination an inspection to verify satisfactory dismantling decontamination and disposal of component parts would be performed by their Regional Compliance Office. The Docket File indicates in a memo to file dated March 5, 1968, that this inspection was performed and found the facility to have been dismantled and all fuel and other radioactive material disposed of elsewhere.

3. Dismantling Order Date

The AEC authorized dismantling on March 10, 1967.

4. Radiological Status Information

<u>Documentation Identifying Primary Contaminant</u>

No information identifying primary contaminants was provided in the Docket File; however, the Facility License authorizes the Licensee to receive, possess and use up to 1200 kilograms of contained U-235 and to possess and use up to 15 curies of polonium encapsulated as a Polonium-Beryllium neutron source and to possess, but not separate, byproduct material produced by operation of the facility.

Applicable Guidelines, Supplemental Limits, Exceptions

Guidelines mentioned in the Docket File are 10 CFR 20 for decontamination, 10 CFR 70 for fuel that was transferred and liquid release rate limits of 3×10^{-8} μCi/cc alpha and 3×10^{-7} μCi/cc beta.

Licensee's Survey

The Docket File does not contain any information on measurement and sampling locations.

Measurement, Sampling and Analyses Procedures

The Docket File does not contain any information on measurement, sampling and analyses procedures.

Final Survey Results

The Docket File does not contain any information on measurement and sampling locations. The application for license termination mentions the inspection of all reactor components for induced and removable radioactivity. It was noted that no detectable surface contamination was present and that a few parts with induced radioactivity were packaged as radioactive waste and transferred to a licensed waste disposal firm. It was also noted that the reactor cell and all other locations used in connection with operation of the MSCA are free of radioactivity.

Confirmatory Survey Results

The AEC's License Termination Order indicates that GE's MSCA has been dismantled and disposition made of component parts. The order also mentions that surveys of component parts indicated that no detectable radioactivity above background was identified, however, it is not clear whether these surveys were conducted by General Electric or the AEC.

The AEC's February 12, 1968, Division of Compliance inspection indicates that the only detectable radioactivity was on the fast core grid and shroud (1000 cpm) and that both were shipped to GE's waste storage yard. This inspection also lists detailed fuel transfer information including recipient, percent enrichment and quantity transferred. The inspector surveyed the Critical Facility cell and vault with portable alpha and beta-gamma survey meters. Smears were also taken on equipment in the cell, fuel storage racks in the vault and on both floors. It was noted that no significant radioactivity above background levels was noted on smears or during surveys. Analysis of dump tank water showed 1.3×10^{-9} μCi/cc alpha and 1.4×10^{-7} μCi/cc beta. No confirmatory radiological data are provided in the Docket File.

5. License Termination Date

The AEC issued a letter for termination of the license on March 11, 1968.

6. Federal Register Notice

The notice of issuance of order authorizing dismantling of the facility was sent to the Federal Register for publication on March 13, 1967.

The Notice of Termination of Facility License was sent to the Federal Register for publication in March 14, 1968.

7. Hazardous Wastes Identified at the Site

No hazardous wastes were indicated in the Docket File.

8. Restrictions/Conditions/Concerns

No restrictions, conditions, or concerns regarding future use of the facility were described in the Docket File.

9. Does the Site Meet the Present Criteria?

Data on total residual surface activity levels, removable contamination and exposure rates was not provided; therefore, it is not possible to evaluate conditions relative to guidelines.

The Docket File does not adequately identify the extent and location of the surveys performed. No drawings, survey maps or procedures are included.

10. Recommendations

An attempt should be made to obtain documentation of survey activities after decontamination and decommissioning were completed. If such documentation is not available a survey would be recommended to accurately assess the radiological condition of the site.

Prepared by: A. T. Payne
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: January 29, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site: General Dynamics Corporation
Address: Torrey Pines Mesa, California
Facility: Fast Critical Assembly Type Nuclear Reactor
Site Contact: B. B. Smyth, Vice President
NRC Contact: Saul Levine, Division of Reactor Licensing
Docket No.: 50-212
License No: R-96

1. Summary of Operating History

On September 22, 1964, General Dynamics Corporation was licensed to operate a critical facility at their Torrey Pines Mesa laboratory site near San Diego, CA site. The Fast Critical Assembly was owned by the AEC and operated by General Dynamics Corporation to carry out work for the Defense Atomic Support Agency.

Studies of transient radiation effects on electronics, materials, components, and circuits were conducted. The facility and processes are outlined in the Docket File. The approved maximum power level was 500 watts (thermal).

2. Request for Termination of the License

A request for termination of license R-96 was submitted to the AEC on February 1, 1965.

3. Dismantling Order Date

A dismantling order by the AEC is not present in the Docket File. A letter from General Dynamics Corporation dated October 16, 1964, stated intentions to suspend assembly operations but requested that the license R-96 remain intact until a replacement license covering only possession of special nuclear material was obtained.

4. Radiological Status Information

<u>Documentation Identifying Primary Contaminant</u>

No information identifying primary contaminants was provided in the Docket File. The License approves the receipt, possession and use of up to 65 kg of contained uranium 235 and possession, but not separation, of by-product material produced by operation of the reactor.

<u>Applicable Guidelines, Supplemental Limits, Exceptions</u>

The Docket File does not identify the applicable guidelines.

<u>Licensee's Survey</u>

The Docket File does not contain any information concerning surveys conducted by the Licensee.

Measurement, Sampling, and Analysis Procedures

The Docket File does not contain any information on measurement, sampling and analysis procedures.

Final Survey Data

The Docket File does not contain any information concerning a final survey conducted by the Licensee. A letter from General Dynamics Corporation to the AEC dated October 16, 1964, indicates that the assembly machine was to be dismantled and packaged in compliance with regulations and delivered to Union Carbide Nuclear Company.

Confirmatory Survey Results

The Docket File does not contain any information concerning confirmatory survey activities.

5. License Termination Date

The AEC issued a Notice of Termination of the license on March 5, 1965.

6. Federal Register Notice

The Docket File contains a letter dated March 9, 1965, requesting publication of the Notice of Termination of Facility License #R-96 which includes a statement indicating that Special Nuclear Materials License No. SNM-862 was issued authorizing possession and storage of the fuel.

7. Hazardous Waste Identified at the Site

No hazardous wastes were identified in the Docket File.

8. Restrictions/Conditions/Concerns

None were noted in the Docket File.

9. Does the Site Meet Present Criteria?

No information is included in the Docket File concerning the final radiological condition of the site, therefore, an assessment cannot be made.

10. Recommendations

An attempt should be made to contact General Dynamics Corporation to request any information they may have concerning the final radiological condition of the site. If this information is not available or does not confirm the status of the facility relative to the present guidelines, a confirmatory survey should be performed. Additional information may be available in files for license SNM-862.orr

Prepared by: Ann Payne
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: January 30, 1991

REVIEW OF TEST AND RESEARCH REACTOR TERMINATED LICENSES

Site:	Polytechnic Institute of New York
Address:	333 Jay St, Brooklyn, NY 11201
Facility:	AGN-201M Research Reactor
Site Contact:	John R. Lamarsh, Dept. Head Nuclear Engineering
NRC Contact:	Karl R. Goller, Asst. Director for Operating Reactors
Docket No.:	50-216
License No:	R-107

1. Summary of Operating History

In 1964, the AGN-201M reactor, Serial Number 105, was transferred from the U.S. Naval Hospital, National Naval Medical Center, Bethesda, MD to New York University. New York University (NYU) placed the reactor in its School of Engineering and Science at University Heights Campus, in the borough of the Bronx in New York City. In 1973, the School of Engineering and Science of New York University merged with the Polytechnic Institute of Brooklyn to form the Polytechnic Institute of New York (PINY). The reactor and license was transferred from NYU to PINY. The facility was used for the instruction of students in the nuclear engineering program.

The PINY AGN-201M reactor was a small research reactor designed to operate at a maximum power of 0.1 watts. The core of the reactor consisted of UO_2 embedded in a polyethylene moderator. The uranium was enriched to 20% U-235. The core was made up of a number of polyethylene/uranium disks. Each was approximately 10 inches in diameter and of varying thickness. A reflector of 20 cm of high density graphite surrounded the core. There were two safety and two control rods. The rods were made of the same UO_2 impregnated polyethylene as used in the core. Thus, reactivity was decreased when the rods were removed from the core.

2. Request for Termination of the License

A request for termination of License R-107 was submitted to the NRC on July 1, 1977. Included in this request was the proposed decommissioning plan.

3. Dismantling Order Date

The dismantling order was approved on September 29, 1976. Dismantling was completed on July 1, 1977.

The reactor fuel was removed from the reactor on February 5, 1976 and shipped to Oak Ridge National Laboratories. The reactor components, with the exception of the core tank, were removed on February 28, 1977. The reactor source was disposed of on May 19,1977. The core tank was removed during the week of June 13, 1977. The facility components which included the reactor tank, concrete blocks, lead, control console and miscellaneous equipment were transferred, as non-radioactive material, to the Institute of Resource Management, Bethesda, MD for warehousing pending ultimate disposition.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

No information identifying primary contaminants was provided in the Docket File.

Applicable Guidelines, Supplemental Limits, Exceptions

The applicable guidelines were Regulatory Guide 1.86.

Licensee's Survey

External dose equivalent rates and contamination smear surveys were conducted on February 28, 1977. The areas surveyed included the reactor, concrete shield blocks, the RaBe neutron source, the neutron source shipping drum and reactor room. The results indicated no radioactivity above background levels.

Radiation and wipe surveys conducted on May 23, 1977, upon removal of the source, also indicated background levels.

Measurement, Sampling, and Analyses Procedures

Sampling consisted of smears, taken over a 100 cm^2 area, counted on a Beckman LS 100 Liquid Scintillation System. Efficiency was measured for various beta and alpha radiation spectra ranging from tritium (efficiency = 25%) to mixed fission products (efficiency = 85%).

Alpha-beta measurements were made using a NMC (2π) Proportional Counter, Model PC-3T. Measurements were made using a standard Tl-204 source of 0.0045 μCi. A 100% efficiency was assumed, and a backscattering factor of 24% was measured which was consistent with the 27% given in the Rad Health Handbook for aluminum backing. Seven readings were taken on each sample.

Surveys for fixed radioactivity were performed using a thin window GM capable of detecting levels as low as 0.01 mR/h.

Gamma measurements were made on an Ortec gamma spectrometer using a 2π geometry. A Co-60 standard was used to determine the efficiency of 11% which was typical for a 1.5" X 1" crystal at that energy. All readings were corrected for geometry and efficiency.

Final Survey Data

On May 23, 1977, wipe tests were performed in the reactor building. Alpha-beta measurements indicated no contamination exceeding 25 $\pm$ 9 dpm. The highest gamma measurement was 127 $\pm$ 82 dpm.

Confirmatory Survey Results

Mr. Greenman of the NRC Office of Inspection and Enforcement inspected the facility on July 12 and July 18, 1977. The inspection consisted of selective examination of procedures and representative records, interviews with personnel, and measurements and observations by the inspector. He verified that dismantling had been completed in accordance with the dismantling order, and conducted radiation and contamination surveys of the reactor facility. Background measurements were made using a Ludlum Analyzer. No environmental levels greater than 2200 cpm were identified. Additionally the reactor labs were surveyed. Smear surveys indicated a background of 1773 cpm. All locations surveyed were at or below background levels. Samples obtained included water from the former heat exchanger, soil from beneath the former reactor location, concrete dust and paint from floors, and water from the laboratory sink trap. All samples indicated background levels.

The NRC determined that the facility met the acceptance criteria for unrestricted use.

Future Use of Facility

The Docket File contains no information regarding the future use of the facility.

5. License Termination Date

The license was terminated on December 21, 1977.

6. Federal Register Notice

The Federal Register notice for termination of the license was item 7590-01 in Federal Register, Volume 43, No. 2, Wednesday, January 4, 1978.

7. Hazardous Waste Identified at the Site

No hazardous wastes were indicated in the Docket File.

8. Restrictions/Conditions/Concerns

The Docket File did not indicate any restrictions, conditions or concerns.

9. Does the Site Meet Present Criteria?

The data presented by the licensee and NRC inspector in the Docket File indicates that the surveys performed meet present criteria. However, the survey data does not indicate the radionuclides of concern and other pertinent information necessary to fully analyze the survey results in relation to the criteria. Additionally, the Docket File does not contain detailed maps indicating survey locations. Therefore, the extent and adequacy of the surveys performed cannot be evaluated.

10. Recommendations

If additional survey data exist, they should be reviewed to determine whether the survey results adequately demonstrate results were below the guidelines. If available, the survey reports should be reviewed to determine if the scope of the surveys performed were sufficient. If it is determined that the scope was not adequate, an abbreviated survey of the facilities may be conducted. However, due to the time elapsed since dismantling occurred and other uses of the facility, it is unlikely that residual contamination from the R-107 license would still be present. Furthermore, if contamination were indicated, it would be very difficult to associate it with the license.

Prepared by: Betty M. Smith
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: January 28, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	General Atomic Company
Address:	P.O. Box 81608, San Diego, CA 92138
Facility:	TRIGA Mark III
Site Contact:	Douglas T. Farney, Licensing Administrator
NRC Contact:	R.W. Reed, Operating Reactors Branch #4
Docket No.:	50-227
License No:	R-100

1. Summary of Operating History

The General Atomic Company TRIGA Mark III reactor was located in the TRIGA Building at the General Atomic Company's John Jay Hopkins Laboratory for Pure and Applied Science in San Diego, California. The TRIGA Mark III reactor was designed primarily for in-core testing of thermionic cells at constant power levels. Facilities were also provided for high-energy neutron and gamma radiation studies and sample activation. These facilities included two beam ports, an isotope production facility surrounding the core (rotary specimen rack), a pneumatic transfer system, a large reactor pool for in-pool irradiations, and facilities for in-core irradiations. The reactor core was operated near the bottom of a large, open, water-filled pool, and was suspended from a moveable bridge. The aluminum lined pool was 15 feet long, 10 feet wide, and 26 feet deep, and was surrounded by a concrete shield. The fuel moderator elements were clad with stainless steel and contained a homogeneous mixture of uranium-zirconium hydride.

Start-up occurred on January 19, 1966. Power operation ceased on August 17, 1973, and all critical operations ceased on November 28, 1973. The TRIGA Mark III reactor was operated for approximately 84 months between January 1966 and January 1973 at an average power of 1.5 Mw(Th). Fast neutron leakage from the reactor core was approximately 1.5 E+07 nv/w.

On November 28, 1973, all reactor fuel, control rods and start-up sources were removed from the Mark III reactor and transferred to the Mark F (License R-67) facility.

There were no accidents or spills of radioactive material resulting in contamination of this facility during its operation.

2. Request for Termination of the License

A request for termination of License R-100 was submitted to the NRC on September 23, 1975 and supplemented by a letter on October 29, 1975 by providing further information on the level of activity in the pool liner and concrete shield which would remain in place after termination of the license.

3. Dismantling Order Date

A dismantling order was approved by the NRC on August 15, 1975. Dismantling was completed on August 29, 1975.

The dismantling plan called for the facility to obtain a license amendment from the State of California for control over the grid plate, core supports and core shrouds; and for the evaluation of the activity of the under water biological shield and pool liner.

As part of the dismantling plan, the pool liner and concrete biological shield were to remain in place after termination of the license, since the facility was to be used in another licensed activity. Calculations were performed to show that the activity induced in the pool liner and biological shield were below levels acceptable for release for unrestricted area use. The activity of the liner was calculated to be approximately 1.2 E-03 μCi/g of Co-60 and the activity of the shield was approximately 3.2 E-12 Ci/g of Na-24. It was decided that the removable activity that was present under water would be below the guideline levels since the water in the pool had been < 17 ρCi/l since reactor operations ceased.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

No information identifying primary contaminants was provided in the Docket File.

Applicable Guidelines, Supplemental Limits, Exceptions

The primary guidelines utilized included Regulatory Guide 1.86 and 10 CFR 20, Appendix B.

Licensee's Survey

In March 1975, surveys indicated that the instrument control, console, and areas above the pool water level (including floor and drains, support platform, and vent ducts) were < 10 dpm β-γ/100 cm^2 and <100 dpm α/100 cm^2 removable, and <100 dpm β-γ/100 cm^2 and <100 dpm α/100 cm^2 fixed.

The water present in the pool was <17 ρCi/l.

Measurement, Sampling, and Analyses Procedures

The Docket File does not contain any information on measurement, sampling, or analysis procedures.

Final Survey Data

Radiation and smear surveys taken subsequent to the dismantling operations show that the radiation levels of accessible surfaces remained below the guideline levels. Twenty-one smear and meter survey measurements were taken. Smears were <5 dpm alpha/100 cm^2 and < 3 dpm beta/100 cm^2. All beta-gamma meter survey results were <0.2 mrem/h. Additionally, a water sample was analyzed and was <1.2 E-08 μCi/ml alpha and < 1.6 E-08 μCi/ml beta.

Confirmatory Survey Results

The facility was inspected by Mr. A.D. Johnson of the Office of Inspection and Enforcement on October 15, 1975 (Inspection Report 50-227/75-02). The inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector. He examined the results of routine wipe and meter surveys conducted by the licensee on September 11, 1975 around the pool area.

Future Use of Facility

The facility structures, including the reactor pool, would be used to conduct laboratory-type operations using Special Nuclear Material (SNM) with some mixed fission products, as licensed under SNM-696 Material License and California By-product License 0145-59.

5. License Termination Date

The NRC issued a letter for termination of the license on December 10, 1975.

6. Federal Register Notice

No Federal Register notices were include in the Docket File.

7. Hazardous Waste Identified at the Site

No hazardous wastes were identified in the Docket File.

8. Restrictions/Conditions/Concerns

None were noted in the Docket File.

9. Does the Site Meet Present Criteria?

The removable contamination levels reported by the licensee satisfy the present criteria. No data on the primary radionuclides, procedures, instrumentation used, areas surveyed, total activity levels and exposure rates are provided in the Docket File. Therefore, assessments of these parameters relative to the guidelines cannot be made.

10. Recommendations

Additional survey and analysis data should be reviewed to determine the status of the facility. Radiological monitoring of the facility should have continued since the facility was to be used under License SNM-696.

Prepared by: Betty M. Smith
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: January 28, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Gulf General Atomic Company
Address:	P.O. Box 81608, San Diego, CA 92138
Facility:	Thermionic Critical Facility
Site Contact:	Douglas T. Farney, Licensing Administrator
NRC Contact:	Donald J. Skovholt, Assistant Director for Operating Reactors
Docket No.:	50-234
License No:	CX-23

1. Summary of Operating History

The Thermionic Critical Facility (TCF) operated by Gulf General Atomic Company was formerly known as the Tungsten Nuclear Rocket Critical Facility operated by General Dynamics Corporation. The reactor was authorized to operate at a maximum steady state power level of 200 watts (thermal) at the site in Torrey Pines Mesa in San Diego, California. It was licensed on May 25, 1965 and was shut down on September 30, 1969. The facility was previously constructed and operated by the General Atomic Division of General Dynamics Corporation under contract with the Atomic Energy Commission for the conduct of the Experimental Beryllium Oxide Reactor critical experiments and was exempt from licensing requirements. No data are present in the Docket File to indicate how long the facility operated prior to licensing.

The reactor was fueled with highly enriched uranium (approximately 93%) and moderated with water. The solid radioactive materials generated during reactor operation were surveyed by the staff health physicist and disposed of according to the applicable regulations. Gaseous wastes were exhausted through filters on top of the building. The filters were capable of removing 99% of particulate material greater than one micron in size.

Uranium foil and rod stock were stored in the Assembly Building storage vault prior to fabrication operations. TCF fabricated various fuel elements composed of different combinations of uranium, moderator, and diluent foils. The cutting and fabrication process took place in the Assembly Building. The complete fuel element subassemblies were stored in the facility vault attached to the Control Building. Complete fuel elements were stored in the Critical Assembly Building.

Upon dismantlement the fuel was shipped to the Union Carbide Corporation (Nuclear Division) in Oak Ridge, Tennessee.

2. Request for Termination of the License

A request for termination of license CX-23 was submitted to the AEC on May 25, 1973.

3. Dismantling Order Date

On February 20, 1973, Gulf General Atomic Company submitted the Dismantling Plan for TCF to the Atomic Energy Commission for approval. The plan called for the disassembly and decontamination of the assembly components; the removal of all reactor components, control instrumentation, and ancillary equipment from the facility to an on site storage area; and the decontamination of the facility to as low as possible levels.

The AEC approved the Dismantling Plan on March 30, 1973.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

According to Special Nuclear Material (SNM) License SNM-69 Appendix D-2, General Atomic was allowed to possess approximately 400 kgs of uranium metal enriched to approximately 93% in U-235, primarily in the form of foils and rod stock.

Applicable Guidelines, Supplemental Limits, Exceptions

The applicable guideline at the time of dismantling was the "AEC Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use, April 22, 1970." This specified that the limit for removable contamination was 100 dpm/100 cm^2 for alpha and 1000 dpm/100 cm^2 for beta-gamma.

Licensee's Survey

The Licensee performed surveys in the control room and reactor building including the floor and equipment.

The Licensee performed a characterization survey prior to dismantlement operations. The survey of the floor and equipment in the control room indicated $<$ 300 dpm/100 cm^2 alpha by direct measurement with a PAC-ISA and from 9 to 22 dpm/100 cm^2 removable alpha, with smears being proportional counted. The total beta-gamma measurements of the floor and equipment in the control room, using a Juno Model 8, indicated $<$ 0.1 mrad/h at 1 cm and the removable beta-gamma ranged from 57 to 213 dpm/100 cm^2, with smears being counted proportionally.

In the reactor building, surveys of the floor and equipment indicated the total alpha activity was $<$ 300 dpm/100 cm^2 by direct measurement and the total beta-gamma activity was $<$ 0.1 mrad/h at 1 cm. The smear surveys of the floor and equipment in the reactor building indicated removable alpha contamination ranged from 13 to 352 dpm/100 cm^2 and removable beta-gamma contamination ranged from 37 to 352 dpm/100 cm^2, with smears being proportional counted.

Smear surveys of the reactor room ventilation indicated 60 dpm/100 cm^2. However, the report does not indicate if this was alpha or beta-gamma contamination.

Measurement, Sampling, and Analyses Procedures

The Docket File does not contain any information on measurement, sampling, and analysis procedures.

Final Survey Data

Wipe surveys in the control room indicated < 10 ρCi/100 cm^2 alpha and < 15 ρCi/100 cm^2 beta-gamma. All smears were proportionally counted. A meter survey indicated all control room areas were < 0.2 mrad/h beta-gamma and < 300 cpm alpha. Wipe surveys of the reactor room indicated a maximum of 36 ρCi/100 cm^2 beta-gamma and a maximum of 27 ρCi/100 cm^2 alpha. Meter surveys in the reactor room indicated < 0.4 mrad/h beta-gamma and < 300 cpm alpha. The Licensee utilized a Juno Model 8 for beta-gamma measurements and an Eberline Model PAC-ISA for alpha measurements.

Confirmatory Survey Results

A closeout inspection was performed by an AEC Region V Inspector. The inspection disclosed that the dismantling of the reactor, and the storage of equipment and instrumentation had been in accordance with the Dismantling Plan. A survey was made during the inspection which included radiation measurements and smear surveys for removable contamination.

A radiation survey was performed in the control room, reactor room, entrance way to the reactor room, storage room, and ventilation control room. The reactor room, entrance way, and control room were surveyed directly for residual alpha contamination using an Eberline Scintillation Counter, Model PAC-ISA that had been calibrated July 1, 1973. All areas were also surveyed directly for beta-gamma residual contamination using a thin-window GM meter that had been calibrated May 21, 1973. No residual alpha or beta-gamma contamination was indicated above background levels.

Smears were taken in all areas as part of the survey. The smears were counted in a Nuclear Measurements Corporation Model PC-3A Proportional Counter. The results indicated alpha plus beta activity was < 50 dpm/100 cm^2 for all smears. The limit for removable contamination in the guidelines was 100 dpm/100 cm^2. The corresponding limit for beta-gamma contamination was 1000 dpm/100 cm^2. The results of the survey showed the Licensee had met the "AEC Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use, April 22, 1970."

The report indicated that elevated radiation levels existed in the reactor control room. The source was an adjoining room used by Iso Med Corporation to process molybdenum-99. The AEC survey did not indicate specific areas of contamination in the control room above the elevated readings. Since the smear results indicated no significant activity, and the fact that no radiation above background or removable contamination were detected in areas other than the control room, the AEC concluded that there was reasonable assurance that the control room met the limits of the guidelines.

Future Use of Facility

It was expected that the continued use of the building would be in areas of nuclear research.

5. License Termination Date

The AEC recommended termination of the license on August 10, 1973.

6. Federal Register Notice

No Federal Register notices were included in the Docket File.

7. Hazardous Waste Identified at the Site

The Docket File contained no information on hazardous wastes.

8. Restrictions/Conditions/Concerns

The Docket File does not indicate any restrictions, conditions, or concerns.

9. Does the Site Meet Present Criteria?

The surveys performed by the Licensee and AEC meet the present criteria of Table 1 of Regulatory Guide 1.86. However, the extent of the surveys performed are not indicated in the Docket File. The information provided indicates that only floors and equipment were surveyed. Walls, upper surfaces, ceilings, pipe runs, etc., were not addressed. The report indicates that there was a potential for contamination of walls and upper surfaces, since contamination was indicated in the reactor room ventilation.

10. Recommendations

A more detailed survey may need to be conducted, if additional survey information is unavailable. The future use of the facility indicates that it would continue to be used for nuclear researach and due to the elapsed time since termination of operations and other uses that have occurred since license termination, it is unlikely that residual contamination from the CX-23 license would still be present. Furthermore, even if any contamination were found, it would be very difficult to associate it with the license.

Prepared by: Betty M. Smith
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: January 29, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Gulf General Atomics
Address:	Torrey Pines Mesa, San Diego, CA
Facility:	Accelerator Pulsed Fast Critical Assembly
Site Contact:	E. Creutz, Vice President Research and Development, General Dynamics
NRC Contact:	P.A. Morris, AEC Division of Reactor Licensing
Docket No.:	50-235
License No.:	R-99

1. Summary of Operating History

License R-99 for operation of the accelerator Pulsed Fast Critical Assembly at up to 500 watts (thermal) was issued June 29, 1965. The Docket File contains no information regarding the design of the facility or the fuel material used. No information concerning spills, incidents, or fuel failures was provided.

2. Request for Termination of the License

The request for license termination, accompanied by a very brief plan for decommissioning, was submitted on September 12, 1969. The plan called for disassembly of the machine and return of the various parts to Oak Ridge National Laboratory and Los Alamos Scientific Laboratory. Fuel would be retained by Gulf General Atomics for use in other experiments; it would be transferred to existing license SNM-69.

3. Dismantling Order Date

The Docket File does not contain AEC approval of the decommissioning plan.

4. Radiological Status Information

<u>Licensee's Survey</u>

The licensee reported that no significant radiation or contamination was detected on equipment surfaces. The survey was performed on October 1, 1969, using a GM detector and Whitman 41 filter paper wipes. GM measurements were less than 2 times the background level of 150 cpm - no information to permit conversion to activity level or exposure rate is included. Smearable contamination was less than 35 dpm/ft^2. beta, and less than 6.6 dpm/ft^2, alpha.

There is no mention of surveys conducted on building surfaces.

Potential contaminants were not identified.

Procedures, specific instruments, survey locations, and details of measurement data were not provided in the Docket File.

Applicable guidelines for the decommissioning were not stated in the Docket File.

<u>Confirmatory Survey</u>

An inspection was conducted on October 6 and 7, 1969 by the AEC Region V Division of Compliance. The inspection report indicates that fuel is in storage, awaiting transfer to license SNM-69, and the assembly machine components are awaiting shipment. The inspection report briefly describes the results of the licensee's survey of machine surfaces and states that levels are low enough for release to unrestricted areas. No confirmatory measurements by the AEC inspector are described.

5. License Termination Date

The AEC recommended that the license be terminated on October 22, 1969.

6. Federal Register Notice

A notice of license termination was published in the Federal Register on October 31, 1969.

7. Hazardous Wastes Identified at the Site

No hazardous waste were indicated in the Docket File.

8. Restrictions/Conditions/Concerns

None identified in the Docket File.

9. Does the Site Meet Present Criteria?

The removable contamination levels determined by the licensee satisfy current criteria. Insufficient information is provided in the Docket File to evaluate the GM instrument measurements and no exposure rate data are included.

Equipment has been returned to the original suppliers.

No measurements are provided for surfaces other than those of the critical assembly machine.

Information is therefore inadequate to determine if the facility satisfies the present release criteria.

10. Recommendations

A survey of the area which housed the "reactor" assembly, conducted by the current occupant, NRC Region V, or an agent of the NRC, would determine whether the radiological status satisfies current guidelines. The difficulty may be that other radiological activities could have resulted in contamination that is not associated with license R-99.

Prepared by: J.D. Berger
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: December 26, 1990

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Gulf Oil Corporation
Address:	P.O. Box 608, San Diego, CA 92112
Facility:	HTGR Critical Facility
Site Contact:	Douglas T. Farney, Licensing Administrator
NRC Contact:	Donald J. Skovholt, Assistant Director for Operating Reactors
Docket No.:	50-240
License No:	R-104

1. Summary of Operating History

The HTGR Critical Facility was owned by Gulf Oil Corporation, formerly known as General Dynamics Corporation; its name was changed to Gulf General Atomic in 1967 and in 1971 changed to Gulf Oil Corpo ation. The facility located at the licensee's site at Torrey Pines Mesa near San Diego, California was licensed on September 8, 1966. Operation of the HTGR was terminated on June 30, 1969.

The reactor was operated to perform required core physics measurements and test progress, worth experiments on prototype HTGR control rods, Doppler coefficient measurements, and to generate flux maps using foil irradiation techniques.

No other information on the operating history, spills, incidents, etc., was provided in the Docket File.

2. Request for Termination of the License

A request for termination of license R-104 was submitted to the AEC on March 15, 1973.

3. Dismantling Order Date

The Dismantling Order was submitted to the AEC on December 19, 1972 and approved on January 24, 1973.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The Docket File does not contain any information on the primary contaminants.

Applicable Guidelines, Supplemental Limits, Exceptions

"AEC Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of License for By-product, Source, or Special Nuclear Material, April 22, 1970"

Licensee's Survey

The components of the HTGR Reactor assembly, control consoles, and associated equipment were disassembled, monitored, and removed to an on site storage area. Direct radiation surveys on these components, taken with a PAC-ISA survey meter, indicated <0.2 mrad/h beta-gamma at 1 cm and <100 cpm alpha/60 cm^2 (which is the detection threshold of the PAC-ISA).

A direct radiation survey of the facility indicated all areas were <0.2 mR/h beta-gamma at 1 cm and <100 cpm alpha/60 cm^2.

A wipe survey was performed which included the reactor room, vault area, office, control room, and other components. The wipes indicated residual contamination was $<$ 30 dpm β/ 100 cm^2 and $<$ 6 dpm α/100 cm^2.

Measurement, Sampling, and Analyses Procedures

The Docket File does not contain any measurement, sampling, or analysis procedures.

Final Survey Data

The facility was decontaminated with the equipment in place. Residual contamination detected by the wipe surveys indicated <3 ρCi α/100 cm^2 and <15 ρCi β/100 cm^2. Wipe surveys included exposed surfaces of the facility and reactor equipment.

Radiation levels were <0.6 mR/h in the facility.

Confirmatory Survey Results

The closeout inspection was performed on March 21, 1973 by R.F. Fish, an AEC inspector. Radiation and smear surveys were performed as part of the inspection.

The radiation survey included the assembly room, locker room, office, control room, and equipment in storage. An Anton Electric Lab Model CD V-700 survey meter with an end window GM detector was used. No radiation levels above 0.2 mR/h at 1 cm were detected.

The assembly room floor, fuel storage vault, waste sink in the corridor, and the contaminated shower were also surveyed with an Eberline Alpha Survey Meter, Model PAC-ISA with a scintillation probe. No alpha activity was detected.

A total of 26 smears were taken in the HTGR facility. Smears were counted in a Nuclear Measurement Corporation Model PC-3A proportional counter. All smears indicated background levels.

A sample of the water and solids contained in the holdup tank were sampled. The results of sample analysis indicated that no uranium was detected, beta activity was 6.5 E-07 μCi/ml and alpha activity was 4.8 E-07 μCi/ml. The AEC concluded that this very slight amount of activity could be diluted with uncontaminated water to acceptable values for release to unrestricted areas.

Future Use of Facility

It was expected that the facility would be used for other nuclear research.

5. License Termination Date

The AEC issued a letter for termination of the license on April 20, 1973.

6. Federal Register Notice

No Federal Register notices are included in the Docket File.

7. Hazardous Waste Identified at the Site

The Docket File contains no information on hazardous wastes at the site.

8. Restrictions/Conditions/Concerns

The Docket File does not identify any restrictions, conditions, or concerns.

9. Does the Site Meet Present Criteria?

The licensee's final survey indicates radiation levels were <0.6 mR/h. It does not indicate if this included background or whether background was subtracted. Hence, it cannot be determined if this meets current criteria. However, the AEC confirmatory survey indicated radiation levels did not exceed 0.2 mR/h.

The wipe surveys performed indicated that residual contamination would meet present criteria.

The Docket File does not adequately identify the extent and location of the surveys performed. It does not indicate if walls and ceilings were surveyed. Hence, it cannot be determined if the final survey adequately demonstrated compliance with criteria.

10. Recommendations

Either more detailed information would need to be provided or a more detailed survey may need to be performed to verify the adequacy of decontamination of the site. However, due to the time elapsed since termination of operations and other uses that have occurred since license termination, it is unlikely that residual contamination from the R-104 license would still be present. Additionally, even if any contamination was found, it would be very difficult to associate it with the license.

Prepared by: Betty M. Smith
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: January 30, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site: General Dynamics Corporation (General Atomics)
Address: Torrey Pines Mesa, San Diego, CA
Facility: Annular Core Reactor Experiment
Site Contact: E. Creutz, Vice President Research and Development, General Dynamics
NRC Contact: P.A. Morris, AEC Division of Reactor Licensing
Docket No.: 50-246
License No.: CX-24

1. Summary of Operating History

License CX-24 for operation of the Annular Core Reactor Experiment at a maximum power level of 10 kilowatts (thermal) was issued February 17, 1966. There is no facility history, including information on spills, incidents, or fuel failures in the Docket File. The facility design is not described in the Docket File.

The facility had a fairly brief lifetime; a request to terminate the license being submitted in late 1966.

2. Request for Termination of the License

General Dynamics Corp. submitted the request for license termination on October 26, 1966, notifying the AEC at that time that the facility had been dismantled and noting that fuel ion chambers, and control rods had been returned to the TRIGA Mark F reactor (license R-67) and the activated grid plates, support structure, and experimental tube would be disposed of as radioactive waste or transferred to Sandia Corporation, if they were interested.

3. Dismantling Order Date

A dismantling order is not included in the Docket File.

4. Radiological Statue Information

<u>Documentation Identifying Primary Contaminant</u>

The license authorizes possession of 10 kilograms of U-235 and 30 grams of Plutonium. Also, two neutron sources (15 Ci PoBe and 150 Ci SbBe) are authorized.

<u>Applicable Guidelines, Supplemental Limits, Exceptions</u>

The Docket File does not list the established criteria for this decommissioning action.

<u>Surveys</u>

No radiological survey data, developed by the licensee or by the Region II inspection on October 25 and 26, 1966, are provided in the Docket File. The documents include no statement as to the acceptability of the facility for release to unrestricted use.

Information as to direct radiation levels on those items awaiting disposition is provided.

5. License Termination Date

The AEC recommended license termination on December 30, 1966.

6. Federal Register Notice

The notice of license termination was published in the Federal Register on January 10, 1967.

7. Hazardous Wastes Identified at the Site

No hazardous waste was indicated in the Docket File.

8. Restriction/Conditions/Concerns

The components with induced activity were transferred to the State of California license 0145-80.

9. Does the Site Meet Present Criteria?

The Docket File does not contain sufficient information to determine whether the facility meets present radiological criteria.

10. Recommendations

A survey of the area which housed the reactor, conducted by the current occupant, NRC Region V, or an agent of the NRC, would determine whether the radiological status satisfies current guidelines. The difficulty may be that other radiological activities could have resulted in contamination that is not associated with license CX-24. Remedial activity was transferred to a State of California license.

Prepared by: J.D. Berger
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: December 26, 1990

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site: Gulf Oil Corporation
Address: Gulf Energy and Environmental Systems, P.O. Box 81608, San Diego, CA 92138
Facility: Accelerator Pulsed Fast Assembly (APFA III)
Site Contact: Douglas T. Farney, Licensing Administrator
NRC Contact: Donald J. Skovholt, Assistant Director for Operating Reactors
Docket No.: 50-253
License No: R-105

1. Summary of Operating History

The Accelerator Pulsed Fast Assembly (APFA III) reactor was located at the licensee's Torrey Pines Mesa laboratory site, near San Diego, California. The APFA III reactor was owned by the AEC, but was on loan to the Defense Nuclear Agency and accountable by Gulf Oil Company under a facilities contract.

No additional information on operating history, spills, accidents or incidents was present in the Docket File.

2. Request for Termination of the License

A request for termination of license R-105 was submitted to the AEC on June 8, 1973. It was noted that one bending magnet and focusing quadrapole were retained in the APFA cell area. These components were used to transport an electron beam through the cell to another experimental area. These components had radiation levels up to 2.5 mR/h which were generated from gamma-neutron interactions produced by an electron beam. Additional activities concerning the bending magnet were to be performed under the authorization of a State of California license.

The control console and reactor controls were to remain in place.

3. Dismantling Order Date

The AEC approved the Dismantling Order on March 8, 1973.

4. Radiological Status Information

<u>Documentation Identifying Primary Contaminant</u>

No information identifying primary contaminants was present in the Docket Files.

<u>Applicable Guidelines, Supplemental Limits, Exceptions</u>

"Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-product, Source, or Special Nuclear Material," April 22, 1970.

Licensee's Survey

On May 3, 1973, the licensee conducted surveys for direct radiation and removable contamination.

Measurement, Sampling, and Analyses Procedures

The Docket File contains no information on measurement, sampling, or analysis procedures.

Final Survey Data

Surveys were conducted on May 3, 1973.

Direct beta-gamma radiation measurements of the APFA control area indicated background levels. Wipe surveys of the control room floor and equipment indicated residual contamination did not exceed 15 ρCi β/ft^2 and 3 ρCi α/ft^2.

The direct radiation meter survey of the APFA reactor cell indicated approximately 0.5 mR/h beta-gamma at 1 inch. A bending magnet and focusing quadrapole attached to the wall of the APFA cell read up to 2.5 mR/h on internal areas. Wipe surveys were performed on cabinets, floors, walls, drains,exhaust stack, and recycling and exhaust ducts. The wipe surveys indicated residual beta contamination ranged from <15 to 108 pCi/ft^2 and residual alpha contamination ranged from <3 to 169 pCi/ft^2.

Direct radiation measurements were also performed on the reactor assembly machine, and indicated <0.2 mR/h beta-gamma at 1 cm. The control rod support rods read up to 0.9 mR/h. Wipe surveys were performed on the control rod supports, shelves, screen, and base plate. Results indicated residual beta contamination ranged from <15 to 104 pCi/ft^2 and residual alpha contamination ranged from 12 to 596 pCi/ft^2.

Confirmatory Survey Results

On July 8, 1973, a closeout inspection was performed by R.F. Fish, a Region V inspector. The inspection included radiation and smear surveys. The surveys indicated that the levels of removable contamination were below the limits specified in the guidelines. The radiation survey indicated elevated radiation levels in the reactor cell. The bending magnet in this area, which contributed to the elevated readings, would continue to be possessed and used under a State of California license. Additionally, a metal plate was found with alpha contamination. This plate was to be removed by the licensee. No beta-gamma radiation levels greater than 0.2 mR/h were detected.

A radiation survey was performed in the reactor cell and control room. Both areas were surveyed directly for alpha contamination using an Eberline Scintillation Type Counter, Model PAC 1SA. All areas were also surveyed directly for beta-gamma contamination using a thin end window GM meter. The only significant alpha contamination detected (2000 cpm) was a small area on a sheet of metal covering the drive motor for the well shielding door. The licensee intended to dispose of it as contaminated waste or use it as a contaminated item in another controlled area. No beta-gamma contamination above background was detected.

Background radiation levels in the reactor cell varied between 0.2 and 4.5 mR/h. These radiation levels were caused by the operation of the linear accelerator, equipment located in an adjoining cell, and a bending magnet stored in the reactor cell.

A total of eleven smears were taken in the reactor cell and control room. The smears were counted in a Nuclear Measurements Corporation Model PC-3A Proportional Counter. The results indicated alpha plus beta activity present on all smears was <25 dpm/100 cm^2. The limit for removable alpha contamination in the guidelines was 100 dpm/100 cm^2 and 1000 dpm/100 cm^2 for beta-gamma contamination.

Future Use of Facility

Gulf Oil Corporation wanted to lease the facility or make it generally available for other experimentation.

5. License Termination Date

The AEC approved termination of license R-105 on August 10, 1973.

6. Federal Register Notice

No Federal Register notices were included in the Docket File.

7. Hazardous Waste Identified at the Site

The Docket File contains no information on hazardous wastes.

8. Restrictions/Conditions/Concerns

The Docket File contains no restrictions, conditions, or concerns.

9. Does the Site Meet Present Criteria?

The removable contamination levels reported by the licensee and AEC satisfy the present criteria. However, there is no information on direct alpha or direct beta-gamma measurements made by the licensee. The AEC did perform some direct measurements which indicated that levels were below the guidelines.

Current criteria specify that "the average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h at 1 cm and 1.0 mrad/h at 1 cm, respectively." Surveys performed by the licensee in the reactor cell indicated beta-gamma radiation levels of 0.5 mR/h at 1 inch, which would exceed the present criteria. However, this area also contained the bending magnet reading 2.5 mR/h and this may have influenced the licensee's readings.

The Docket File does not adequately identify the extent and location of the surveys performed. No drawings, maps or procedures are included. Based on the information available it appears that the facility did meet the release criteria, but the thoroughness of the survey cannot be ascertained.

10. Recommendations

The licensee should provide additional information concerning the surveys, if available, to determine if the surveys performed were adequate. The licensee should identify the status of the bending magnet and linear accelerator, and obtain exposure rate data. Additional surveys could be performed, but due to the time elapsed since the termination of operations and other uses of radioactive materials that may have occurred since license termination, it is unlikely that residual contamination from the R-105 license would still be present. Even if any contamination were found, it would be difficult to associate it with the license.

Prepared by: Betty M. Smith
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: January 29, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site: United Nuclear Corporation
Address: Pawling, NY
Facility: Proof Test Facility
Site Contact: R.E. Kropp, Licensing Manager, Gulf United Nuclear
NRC Contact: D.J. Skovholt, AEC Division of Reactor Licensing
Docket No.: 50-290
License No.: CX-25

1. Summary of Operating History

The license to operate a utilization facility at up to 100 watts (thermal) was issued on December 2, 1967. No information on facility operating history is provided in the Docket File. There is no description of the facility design or of incidents, spills, or fuel failures which occurred during operation.

2. Request for Termination of the License

Intent to remove the reactor from operation but not decommission was submitted to the AEC on August 2, 1973. The plan was to remove the special nuclear material from the facility, remove the moderator, and conduct a radiation survey by October 1, 1973.

On October 23, 1973, a request for license termination was submitted, followed by two supplements on November 1, 1973 and March 13, 1974. These documents described the decommissioning plan.

3. Dismantling Order Date

The AEC approved the Dismantling Plan on January 25, 1974.

4. Radiological Status Information

Documentation Identifying Primary, Contaminant

The license authorizes possession of 150 kilograms of U-235, 24 kilograms of plutonium, and a 5 Ci PoBe neutron source. Uranium, plutonium, and mixed activation and fission products would be the expected potential contaminants at this site.

Applicable Guideline, Supplemental Limits, Exception

The applicable guideline at the time of dismantling was the "AEC Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use, April 22, 1970."

Licensee's Survey

A predecontamination survey, using a Texas Nuclear Model 2650 survey meter, identified direct gamma radiation levels of 0.12 to 0.15 mR/h; these levels were attributed to the presence of bundles of fuel in the vicinity of the reactor facility. Smears of reactor components indicated a maximum of 1 dpm/100 cm^2, alpha and 19 dpm/100 cm^2, beta-gamma, removable contamination. Rooms in the building had removable contamination levels of 0 to 7 dpm/100 cm^2, alpha, and 0-5 dpm/100

cm^2 beta-gamma. No information regarding specific survey locations is provided in the Docket File.

A post decontamination survey was performed, (date not given but between January 25 and March 13, 1974) using the following Eberline instruments: PAC-4G, PAC-4S, E-120 with HP-190 probe. Filter papers were counted on a Nuclear Chicago Model 1150 gas proportional counter. A description of the survey procedures is not provided. Some specific survey data are provided for components and building surfaces. Maps or drawings are not included. The survey results indicate less than 200 alpha dpm/100 cm^2 and less the 0.02 beta-gamma mrad/h by direct measurement. Removable activity on components ranged from 0 to 6 alpha and 2 to 18 beta-gamma dpm/100 cm^2 ; on building surfaces the removable levels were 0 to 22 alpha and 2 to 31 beta-gamma dpm/100 cm^2.

Confirmatory Survey

An inspection by AEC Region I on January 10 and 11, 1974, verified that fuel had been removed. There were no radiation measurement data and no statement as to status of the facility, relative to guidelines.

No confirmatory measurements or inspections were conducted after completion of the licensee's final survey.

5. License Termination Date

The AEC recommended that the license be terminated on June 25, 1974.

6. Federal Register Notice

The Docket File does not contain a Federal Register Notice of license termination.

7. Hazardous Wastes Identified at the Site

No hazardous waste was indicated in the Docket File.

8. Restrictions/Conditions/Concerns

None noted in the Docket File.

9. Does the Site Meet Present Criteria?

The final survey results of the licensee indicate that the site meets the current criteria, with the exception of the direct alpha activity level, for which the measurement procedure had a sensitivity limit of 200 dpm/100 cm^2, as compared to the average limit for plutonium of 100 dpm/100 cm^2. It is, therefore, not possible to make a judgment about the alpha activity level from the licensee's data.

A survey of the building was performed in 1987 by Oak Ridge Associated Universities, for the Department of Interior - the current property owner. The findings of that survey demonstrated that the radiological status of the building satisfied the current criteria for release for unrestricted use.

10. Recommendations

No actions required.

Prepared by: J.D. Berger
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: December 26, 1990

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Michigan State University
Address:	College of Engineering, East Lansing, Michigan, 48824-1226
Facility:	TRIGA Reactor
Site Contact:	Bruce Wilkinson, Reactor Supervisor
NRC Contact:	Theodore S. Michaels, Senior Project Leader
Docket No.:	50-294
License No:	R-114

1. Summary of Operating History

Between March 1969 and October 1987, Michigan State University operated a 250 kilowatt (thermal) TRIGA Mark I reactor under AEC/NRC license R-114, for purposes of training students in reactor operation principles and to provide a source of ionizing radiation and neutrons for various research programs. The reactor was fueled with 2.5 kg of 19% enriched uranium-zirconium-hydride. The core was contained in a water filled tank, located approximately 7 m below grade. No major incidents, involving releases of radiological material, are known to have occurred during the operating history of the reactor.

2. Request for Termination of the License

The application for termination of the license was submitted on January 20, 1989, and supplemented on May 4, 1989.

3. Dismantling Order Date

The dismantling Order was approved by the NRC on July 11, 1989.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The Docket File contains no information identifying the primary contaminants.

Applicable Guidelines, Supplemental Limits, Exceptions

Regulatory Guide 1.86 and 5 μR/h above background measured at 1 meter above the surface.

Licensee's Survey

Prior to decontamination operations, a baseline radiological survey was performed to fully characterize the initial radiological conditions within the reactor facility.

During dismantlement operations, only minor levels of surface contamination were encountered.

Water in the reactor tank was agitated to ensure a uniform suspension of particles and a sample was collected. The analysis of the sample indicated activity levels below the release criteria of 10 CFR 20.

Measurement, Sampling, and Analyses Procedures

Detailed procedures were included in the final report and in the Chem Nuclear Systems Inc report, "Radiological Control Procedures for Field Projects (CNSI FS-RP-001)."

Final Survey Data

The final survey included Room 184 Laboratory, Reactor Room, Reactor Pit, Material Storage Pit Area, Control Room, Room 190 Office, and Room 192 Health Physics Laboratory. All areas were scanned for beta-gamma contamination and indicated a maximum of 527 dpm/19 cm^2 (2774 dpm/100 cm^2). All areas were also scanned for gamma radiation using a NaI scintillation detectors and indicated a maximum level of 5 μR/h. Smear surveys were also performed and indicated a maximum of 172 dpm/100 cm^2 beta-gamma and 188 dpm/100 cm^2 alpha. Gamma exposure rate measurements, performed with a Reuter-Stokes PIC, indicated a maximum of 3 μR/h inside the Reactor Pit.

Confirmatory Survey Results

A confirmatory survey was performed by a NRC contractor, Oak Ridge Associated Universities (ORAU), on December 18, 1989. The survey included surface alpha, beta-gamma, and gamma scans; measurement of direct and removable contamination levels; exposure rate measurements; and determination of radionuclide concentrations in soil and concrete samples. The findings supported the closeout survey performed by the licensee, and confirmed that the radiological conditions of the facility satisfy the NRC guidelines established for release for unrestricted use.

Future Use of Facility

The Docket File does not contain any information concerning the future use of the facility.

5. License Termination Date

The NRC approved termination of license R-114 on April 5, 1990.

6. Federal Register Notice

No Federal Register notices were included in the Docket File.

7. Hazardous Waste Identified at the Site

No hazardous wastes were identified in the Docket File.

8. Restrictions/Conditions/Concerns

The NRC expressed several concerns regarding the licensee's initial Dismantling Plan. The licensee provided additional information which addressed these concerns.

9. Does the Site Meet Present Criteria?

Based on the surveys performed by the licensee and the confirmatory surveys performed by ORAU, the site does meet present criteria.

10. Recommendations

None.

Prepared by: Betty M. Smith
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: February 1, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	NUMEC and Commonwealth of Pennsylvania
Address:	Quehanna, PA
Facility:	Curtiss Wright Nuclear Research Laboratory
Site Contact:	E.K. Reitler, NUMEC; M.K. Goodard, Dept. of Forests and Waters, Commonwealth of PA
NRC Contact:	D.J. Skovholt, AEC Division of Reactor Licensing
Docket No.:	50-310
License No.:	R-72

1. Summary of Operating History

The reactor at the Quehanna site was originally licensed to Curtiss-Wright; the initial license information is not provided in the Docket File. The facility license was transferred to Pennsylvania State University (Docket File 50-174) in 1960. It was not operated as a critical facility after that transfer. On November 9, 1967, the license was again transferred - this time jointly to the Commonwealth of Pennsylvania's Department of Forests and Waters, which held legal title, and Nuclear Materials and Equipment Corporation (NUMEC) for possession only. Reactor fuel and many components had been removed and the shield was being used as a Co-60 irradiator pool under byproduct material license 37-12307-02.

2. Request for Termination of the License

A request for license termination was submitted by NUMEC on January 4, 1971. That request noted that the facility had been further dismantled by transfer of the PuBe source, two fission Chambers, and the reactor core grid to Pennsylvania State University (per January 15, 1969 license amendment), and 5.2 kilograms of U-235 in fuel elements had been transferred to Brookhaven National Laboratory (per March 5, 1969 amendment). Four safety shims containing induced activity were in storage, awaiting burial or transfer to an authorized recipient.

3. Dismantling order Date

No AEC approval of the dismantling plan was included in the Docket File.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The licensee identified the following radionuclides as potential contaminants: Co-60, Sn-90, Fe-55, Fe-59, Ni-59, Ni-63, Co-57, Zn-65, C-14, S-35, Cd-109, Cd-113 m. The Co-60 contamination is attributed to other (non-reactor) activities at the site under the By Product license.

Applicable Guidelines, Supplemental Limits, Exceptions

There was no statement in the Docket File identifying applicable decommissioning criteria.

Licensee's Survey

The licensee provides radiation level data on components that are being held awaiting disposal. These include safety shims, core access elements, reflector elements, and mock-up fuel rods. It is also noted that beta-gamma contamination is present in processing areas of the facility as a result of the other licensed (37-12307-02) activities ongoing at the site. No data regarding radiological conditions of the building or components not destined for disposal are provided in the Docket File.

Confirmatory Survey

AEC Region I Division of Compliance inspected the facility on September 12, 1971. It was noted that the facility was dismantled and components decontaminated or disposed of. No confirmatory radiation measurements were performed.

5. License Termination Date

The AEC recommended that the license be terminated on December 2, 1971.

6. Federal Register Notice

Notice of license termination was published in the Federal Register on December 14, 1971.

7. Hazardous Wastes Identified at the Site

No hazardous wastes were indicated in the Docket File.

8. Restrictions/Conditions/concerns

Residual contamination to be transferred to By-Product license 37-12307-02.

Does the Site Meet Present Criteria?

Although the Docket File does not specifically state whether or not the facility meets the present criteria, references to areas of existing contamination from other site activities suggest that the facility did not satisfy the criteria for unrestricted release.

10. Recommendations

Because residual contamination was transferred to another, continuing license, the final site release would be conducted for that license.

No action needed for closure of the Docket File.

Prepared by: J.D. Berger
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: December 26,1990

REVIEW OF TEST AND RESEARCH REACTOR TERMINATED LICENSES

Site:	Rockwell International
Address:	Rocketdyne Division, 6633 Canoga Ave, Canoga Park, Ca 91304
Facility:	Research Reactor L-85
Site Contact:	M.E. Remley, Director of Nuclear Safety and Licensing
NRC Contact:	Harold Denton, Project Manager
Docket No.:	50-375
License No:	R-118

1. Summary of Operating History

The L-85 Research Reactor, located in Building T093, at the Rockwell International Santa Susana Field Laboratories, was a NRC licensed operating facility since January 5, 1972. From 1952 until 1972, it was an AEC owned facility. The L-85 reactor was initially located at Downey, California, under the designation WBNS (Water Boiling Neutron Source) from 1952 until 1956, where it was operated at a maximum power level of 0.5 watt. It was moved to the Santa Susana Field Laboratories in 1956, modified to increase power to 3 kilowatts and redesignated the AE-6 Reactor. After transfer of ownership from the U.S. Government to Rockwell International and licensing by the NRC, it was operated in support of commercial programs until February 29, 1980.

2. Request for Termination of the License

A request for termination of the license was submitted on March 10, 1980.

3. Dismantling Order Date

The application for a Dismantling Order and the Dismantling plan were submitted on March 10, 1980. The Dismantling Order was approved on February 22, 1983.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The docket file contains no information identifying the primary contaminants.

Applicable Guidelines, Supplemental Limits, Exceptions

Regulatory Guide 1.86 and 5 μR/h above background measured at 1 meter above the surface.

Licensee's Survey

Initially, the premises to be released included Buildings T093, T083, T074, and T453 (see concerns below). However, later it was determined to be only T093.

Measurement, Sampling, and Analyses Procedures

The Dismantling Plan provides information on measurement, sampling, and analysis procedures.

Final Survey Data

The final radiation survey was submitted to the NRC on March 6, 1986. The licensee stated that the facility met the criteria established in the guidelines (see concerns below).

The licensee surveyed 11% of reactor related areas and 5% of non-reactor related areas and the reactor room ceiling (see concerns below).

Areas were scanned for contamination with a portable Ludlum Model 2220 ESG Scaler with a Model 43-1 alpha detector, or a Ludlum Model 44-9 or Technical Associates Model P-11 beta detector. The maximum activity detected in the reactor buildings by the alpha scan was 63 dpm/100 cm^2 and by the beta scan was 3102 dpm/100 cm^2 (see concerns below). The maximum activity detected in the non-reactor buildings by the alpha scans was 17.2 dpm/100 cm^2 and by the beta scans was 1987 dpm/100 cm^2.

Smear surveys were performed using Whatman 540 filter paper wiped over an area of 100 cm^2. Smear activity was counted using a gas flow proportional counter (NMC Model ACS-77 or equivalent). No removable alpha activity was detected and the maximum removable beta activity detected was 98 dpm/100 cm^2. The maximum removable activity detected in the non-reactor buildings was 4 dpm α/100 cm^2 and 93 dpm β/100 cm^2.

Five soil samples were collected around the facility. All soil samples indicated background concentrations of radioactivity.

The ambient exposure rate, as measured by NaI scintillation detectors, showed a maximum in the reactor room of 21.3 μR/h with a mean of 13.9 R/h. Hence, the licensee determined that the release criteria was 5 μR/h above 13.9 μR/h and all areas were within this limit. The maximum ambient radiation exposure rate in the non-reactor buildings was 23.1 μR/h. These elevated levels were attributed to radioactive material stored nearby in the Radioactive Material Disposal Facility. On January 23, 1986 the reactor room was resurveyed due to NRC concerns over the methods employed (see concerns below). Scintillation detectors were cross checked with a Reuter-Stokes Pressurized Ion Chamber. Then the reactor room was resurveyed with the NaI detectors. Also, in response to NRC concerns the "natural background" measurement was revised to be the mean of the measurements in the reactor room (12.5 μR/h). Hence, four locations now exceeded the 5 μR/h above background limit. The licensee determined it would not be desirable to remediate these areas, and so made assumptions and calculations to show that a person working in the area would receive less than 10 mrem/year. In response to further NRC concerns, the areas were resurveyed using a Reuter-Stokes RSS-111 and indicated a maximum level of 18.9 μR/h, and natural background was determined by performing measurements in a nearby building of similar construction and age (11.9 μR/h). Hence, it was determined that only one area, a hole from which activated concrete had been removed, exceeded the guidelines It was proposed by the licensee to cover this area of elevated exposure with concrete to reduce exposure and restore the floor. On February 3, 1987 the licensee submitted the results of additional measurements performed after concrete was poured and surveys indicated that the maximum radiation exposure rate was <5 μR/h above background.

Confirmatory Survey Results

During September 30 through October 2, 1986, the NRC conducted a closeout inspection. Additionally, a closeout survey was performed by a NRC contractor, Oak Ridge Associated Universities (ORAU), from September 30 through October 31, 1986.

The measurements performed by ORAU confirmed the findings described in the Revision A of the licensee's Final Survey Report. Additionally, the February 3, 1987 submittal of "Summary Report of Ambient Exposure Rate Measurements at the L-85 Research Reactor Facility" was reviewed and results indicated that gamma radiation levels were <5 μR/h above background.

The ORAU survey included scans of the floors and lower walls with alpha, beta and gamma detectors; direct measurements of total and removable alpha and beta-gamma contamination levels on random floor, walls, ceilings, ledges and other horizontal surfaces; gamma exposure rate measurements at 1 meter above the floor at random locations and areas of elevated gamma radiation levels identified by the surface scans; in situ gamma spectrums at each location where gamma exposure rate measurements were made; walkover surface scans of outdoor areas with gamma scintillation detectors; gamma spectrum analysis of surface soil samples collected in areas adjacent to building T093; radiation exposure rate and gamma scintillation measurements of building T453 to establish background levels; and gamma exposure rate measurements and gamma spectrum of soil samples collected from four locations surrounding the Santa Susana Field Laboratory to establish background levels. The surveys indicated that the total residual contamination met the criteria for unrestricted use.

Future Use of Facility

Unrestricted access

5. License Termination Date

The NRC approved termination of the license on April 4, 1987.

6. Federal Register Notice

No Federal Register notices were included in the Docket File.

7. Hazardous Waste Identified at the Site

No hazardous wastes were identified in the Docket File.

8. Restrictions/Conditions/Concerns

Several concerns were expressed by the NRC during the closeout inspection. The licensee's final termination survey indicated that the areas to be released included Buildings T093, T083, T074, and T453. However, the NRC license only covered Buildings T093 and T453. They asked the licensee to clarify the facilities used under license R-118. The licensee's response indicated that only Building T093 was used under authorization of license R-118.

The results of the beta contamination scan indicated a maximum value of 3102 dpm/100 cm^2 and an "Inspection Test Statistic" value of 1000.2 dpm/100 cm^2. However, these values exceeded the guidelines for Sr-90. The NRC requested Rockwell International to demonstrate that Sr-90 was not the source of the beta radiation in order to justify the limits used. The licensee proposed to sample the areas of elevated activity and analyze the samples for Sr-90 and other radionuclides. They proposed that if Sr-90 activity exceeded 5% of the total activity, the areas would be decontaminated further. Analysis of the samples showed less than 1.5% of the beta activity was due to Sr-90.

The report stated that the average value of measurements made in the reactor room was used for a "natural background" measurement. However, the NRC determined this result would be biased due to possible existing contamination. The licensee stated that "natural background" would be determined by measurements in a nearby building of similar construction and age.

The NRC expressed concern that surveying 11% of the surfaces was not sufficient to demonstrate that the release criteria were met. They asked Rockwell International to justify the small area surveyed. The licensee provided no justification for the small area surveyed, but decided to rely on the confirmatory survey to provide assurance that the criteria were met.

The NRC expressed concern that the initial survey of the reactor room relied on using a NaI detector, which does not provide a true dose rate measurement. Hence, the licensee made a comparison with a PIC and resurveyed the area with the NaI detectors. However, the NRC expressed further concerns that their comparisons with the PIC measurement only provided a one point cross-check. The licensee agreed to resurvey the reactor room using a Reuter-Stokes RSS-111 PIC.

The licensee proposed to cover areas exhibiting dose rates above 5 μR/h with at least 15 inches of concrete to reduce the dose rates to acceptable levels. The NRC determined that these measurements would need to be confirmed. The licensee agreed to provide the measurements when the areas were covered.

9. Does the Site Meet Present Criteria?

Based on the licensee's survey and the inspection report which discusses the confirmatory survey, the site does meet present criteria.

10. Recommendations

The Docket File is lacking some information to fully assess the adequacy of the surveys. The licensee's letter dated February 3, 1987, which discusses the final surveys performed, is not in the Docket File. Additionally, the ORAU confirmatory survey is not present, it is only summarized. These items should be included in the docket to provide a more thorough documentation of the adequacy of the decontamination and surveys performed.

Prepared by: Betty M. Smith
Oak Ridge Associated Universities
Oak Ridge, TN 37830
Date: January 31, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	California Polytechnic State University
Address:	San Luis Obispo, California 93407
Facility:	AGN-201 Training Reactor
Site Contact:	Thomas A. Schell, Reactor Safety Officer
NRC Contact:	Peter B. Erickson, Project Manager, Operating Reactors Branch #5
Docket No.:	50-394
License No:	R-121

1. Summary of Operating History

The AGN-201 Training Reactor was a small research reactor which operated at a power level of 0.1 watt at California Polytechnic State University (CPSU). The AGN-201 Training Reactor was first operated by the U.S. Naval Postgraduate School in Monterey, California, from April 30, 1957 until 1971. It operated intermittently at a 0.1 watt thermal power level until the NRC authorized a power level of 10 watts thermal in 1963 following facility modifications. A total of 631 watt-days accumulated during operation of this reactor at the USN Postgraduate School. The reactor was shutdown in 1971, dismantled in 1973, and relicensed at a power level of 0.1 watt thermal at CPSU on October 7, 1975. The facility operated periodically until finally shutdown on May 30, 1978. The reactor accumulated 0.043 watt-days of operation at CPSU.

The AGN-201 reactor consisted of two basic units, the reactor unit and the control console. The reactor unit consisted of the reactor core surrounded by a graphite reflector, which in turn was enclosed by a lead and water shield. The control and safety rods were installed vertically in the bottom of the reactor unit and passed through the shield and graphite reflector into the uranium-polyethylene core. The core consisted of a series of disks formed from a mix of polyethylene and UO_2.

The reactor fuel was shipped to the Department of Energy Y-12 Facility in Oak Ridge, Tennessee. The contaminated components were shipped to U.S. Ecology Low Level Radioactive Waste Disposal Facility, Richland, Washington.

2. Request for Termination of the License

A request for termination of license R-121 was submitted to the NRC on April 30, 1981 and supplemented on September 8, 1981 and January 30, 1985.

3. Dismantling Order Date

A request for a Dismantling Order and the proposed Dismantling Plan was submitted on September 8, 1981 and approved October 6, 1981.

4. Radiological Status Information

<u>Documentation Identifying Primary Contaminant</u>

The "Final Decommissioning Report for the AGN-201 Reactor" included an attachment, "Isotopic Composition, Amounts and Disposal Plan for Residual

Activity of AGN-201 Reactor" which identified the primary contaminants. The primary contaminants consisted of Eu-152, Co-60, and Cs-137.

Applicable Guidelines, Supplemental Limits, Exceptions

Regulatory Guide 1.86

Licensee's Survey

A pre-dismantling post-fuel removal survey was conducted on July 20, 1983.

Gamma radiation surveys were conducted using Ludlum Model 125 micro-R meters, and levels ranging from 6 to 35 μR/h (background ranged from 8 to 10 μR/h) were measured.

A smear survey was performed; smears were counted using an Eberline RASCAL PRS-1 with an alpha/beta probe. Background was 46 $\pm$ 2 cpm. The activity levels on the smears ranged from 42 to 69 cpm/100 cm^2 (20 - 230 dpm/100 cm^2).

Measurement, Sampling, and Analyses Procedures

The Dismantling Plan and final report describe in detail all procedures used.

Final Survey Data

Following completion of the dismantling operations, the licensee performed detailed surveys.

The site was surveyed for fixed and removable contamination. Fixed contamination was measured using a Ludlum micro-R meter and an Eberline PRS-1 with a HP-210 pancake GM detector, and an AC-3 alpha scintillation probe. Measurements were taken at contact or a few millimeters from the surface. Gamma readings were 9 $\pm$ 3 μR/h with the micro-R meter, and not detectable above background for the beta and alpha measurements.

Removable contamination was measured with a Harshaw gas flow proportional counter. A total of 3,147 smear samples were analyzed. With the exception of one smear on a hex screw washer, all other measurement results were not discernable from background. The contaminated washer was transferred to the State license and was to be disposed of as radioactive waste.

Based on these measurements the licensee concluded that there was no residual contamination or radioactive materials remaining in the facility.

Confirmatory Survey Results

The closeout inspection was performed by E. M. Garcia and M. Cillis, NRC Region V inspectors, from February 4 through 8, 1985. The inspection consisted of examination of representative records, interviews with personnel, observations by the inspectors, and performance of radiation and contamination surveys.

The inspectors collected a total of 100 smears from areas and objects in the reactor room, the control room, and the nuclear laboratory, which constituted the restricted area of the facility. These samples were analyzed for gross alpha and beta contamination on the NRC's Tennelec LB-5100 thin window gas proportional counter system. The instrument's efficiency for Pu-239 (32%) and Tc-99 (34%) were used in determining the activity. At the 95% confidence level, the highest activity identified was 3 dpm/100 cm^2 alpha and 9 dpm/100 cm^2 beta above background. These values were well below the guideline levels.

Two water samples were collected where water had either condensed or seeped in from the outside. These samples were analyzed using the NRC's Nuclear Data ND-6620 computerized MCA. No licensed radioactive materials were identified.

The inspectors surveyed the facility for fixed alpha, beta, and gamma contamination. Random samples from floors, walls, and major components were surveyed with an Eberline PRM-6 with an AC-3 detector for alpha, an Eberline E-520 with a HP-260 detector for beta, and an Eberline PRM-7 with a NaI Scintillation detector for gamma activity. These measurements identified no residual activity above background.

To determine if any licensed gamma emitting nuclides remained in the facility, the inspectors gathered four in situ gamma spectrums. In association with these spectrums, precise measurements of the gamma dose rates, at a meter from the surface, were made using a Reuter-Stokes Pressurized Ion Chamber, Model RSS-111. A fifth spectrum, and gamma dose rate measurement, was collected in an area of the building that had not been used for licensed activities. The gamma spectrums were collected using the NRC's portable intrinsic germanium detector and Nuclear Data ND-6 portable multichannel analyzer. The spectrums were used for qualitative determination of the nuclides present. These measurements identified a number of naturally occurring radionuclides, but, with one exception, no licensed radioactive materials were identified. The one exception was the presence of cobalt-60 in the spectrum of the reactor tank. However, the Co-60 does not constitute a significant contributor to personnel exposure.

Future Use of Facility

The Aeronautical and Mechanical Engineering Department intended to use the space for dynamometer stands and general engineering laboratory experiments.

5. License Termination Date

The NRC approved license termination on July 19, 1985.

6. Federal Register Notice

No Federal Register notices were included in the Docket File.

7. Hazardous Waste Identified at the Site

No hazardous wastes were identified at the site.

8. Restrictions/Conditions/Concerns

No restrictions, conditions, or concerns were identified in the Docket File.

9. Does the Site Meet Present Criteria?

Yes, the detailed surveys performed by the licensee and the NRC demonstrate that the site does meet present criteria.

10. Recommendations

The detailed information contained in this report was excellent. It provided enough information to adequately evaluate the site. No further action required.

Prepared by: Betty M. Smith
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Oak Ridge, TN 37830

Date: January 31, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site: University of California, Santa Barbara
Address: University of California, Santa Barbara, 93106
Facility: L-77 Training Reactor
Site Contact: A. Edward Profio, Reactor Director
NRC Contact: Theodore S. Michaels, NRC Project Manager
Docket No.: 50-433
License No: R-124

1. Summary of Operating History

The L-77 Training Reactor, located in Broida Hall (Building 572) of the University of California, Santa Barbara, California (UCSB), was a NRC licensed facility (License R-124). The reactor was originally installed and operated at the University of Nevada, Reno, Nevada, and was moved to UCSB in 1974. UCSB received a Facility Operating License on December 3, 1974, and was authorized for start-up in January 1975. The maximum operating power was 10 watts (thermal). The reactor was operated for instructional purposes and activation analyses.

The L-77 was a homogeneous aqueous solution research reactor. The fuel solution was enriched uranyl sulfate dissolved in water, contained in the Reactor Core Tank, which was contained in an inner shield tank.

UCSB submitted a decommissioning plan to the NRC and a dismantling order was issued on August 26, 1986.

2. Request for Termination of the License

The application for termination of the license was submitted to the NRC on September 9, 1985, and supplemented on November 20, and December 9, 1985 and March 24, and June 27, 1986.

3. Dismantling Order Date

The application for decommissioning and the Decommissioning Plan, prepared by Rockwell International Corporation, were submitted to the NRC on August 12, 1985. The dismantling Order was approved on August 26, 1986.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The Docket File does not identify the primary contaminants. However, it was noted that tritium contamination existed near the accelerator in the south end of Room 1251, due to leakage of coolant water from a tritiated target assembly.

Applicable Guidelines, Supplemental Limits, Exceptions

Regulatory Guide 1.86 and 5 μR/h above background at 1 meter.

Licensee's Survey

Components of the L-77 Training Reactor were surveyed for fixed and removable contamination as they were removed. One plug pin had detectable removable contamination. It was wrapped and labelled and added to other core components that were fitted into the box with the core.

The facility survey was concentrated in the center third of the building where the reactor had been located. A Reuter-Stokes pressurized ion chamber (PIC) survey indicated no radiation levels above background at 1 meter above the floor. A Ludlum micro R meter survey of the roof showed no radiation levels above ambient. Swipe surveys found one area of contamination which was determined to be tritium. This area was decontaminated and resurveyed.

Measurement, Sampling, and Analyses Procedures

The licensee's Decommissioning Plan and final survey report provide detailed descriptions of the measurement, sampling, and analysis procedures utilized.

Final Survey Data

The final survey was performed by the decommissioning contractor, Rockwell International, in September 1986.

Total surface contamination of the entire available floor and middle walls was measured by alpha scintillation and thin-window GM probes and determined to be <10 dpm alpha/100 cm^2 and <1250 dpm beta-gamma/100 cm^2. Since the limit for Sr-90 activity was 1000 dpm/100 cm^2, additional calculations and surveys were performed to confirm that Sr-90 was not present.

Removable surface contamination of the floor and middle walls was assessed on an alpha/beta gas-flow proportional counter. One smear showed detectable activity. Additional smears were taken and counted on a liquid scintillation spectrometer. From this set, four had detectable activity, with a beta emitter in the energy range of tritium. The areas involved were decontaminated and resurveyed. Final surveys indicated background activity levels.

Four water samples were analyzed by gamma spectrometry using a HPGe spectrometer with 500 ml samples. None indicated the presence of radioactivity. Ambient radiation exposure rates were measured using a Reuter-Stokes RSS-111 PIC and Ludlum micro R meter at 50 and 200 points, respectively. Additional Ludlum micro R meter measurements were performed on the roof and around the building. All measurements indicated background levels.

Confirmatory Survey Results

A NRC contractor, Oak Ridge Associated Universities (ORAU) performed a confirmatory survey on June 14-15, 1989. The report was submitted to the NRC in September 1989.

Systematic alpha, beta-gamma, and gamma scans were performed on floors and lower walls (up to 6 feet) using gas proportional alpha/beta floor monitors, zinc sulfide alpha detectors, pancake GM detectors, and NaI(Tl) scintillation detectors coupled to scalers/ratemeters with audible indicators. No areas of elevated contact radiation were detected.

Thirty-five grid blocks on the floor and lower walls were randomly selected for surface contamination measurements. Total measurements of alpha and beta-gamma contamination levels were systematically performed at the center and at four points, midway between the center and the block corners. Smears for removable alpha and beta contamination were performed at the location in each grid block where the highest direct reading was obtained. Total and removable contamination levels were also measured at 20 locations on the upper walls, ceilings, and miscellaneous overhead objects. The maximum alpha measurement was 100 dpm/100 cm^2 and the maximum beta-gamma measurement was 1600 dpm/100 cm^2. The highest alpha grid block average was 56 dpm/100 cm^2 and the highest beta-gamma grid block average was 1200 dpm/100 cm^2. The maximum removable alpha, beta, and tritium activity levels were 5 dpm/100 cm^2, 14 dpm/100 cm^2, and 9 dpm/100 cm^2, respectively.

Gamma exposure rates at 1 meter above the floor were measured at 6 locations using a pressurized ion chamber. The maximum level measured was 12 μR/h. Baseline measurements ranged from 9 to 11 μR/h.

In-situ gamma spectra were collected at several locations where gamma exposure rate measurements were made. The spectra only identified the presence of natural radionuclides normally present in building materials.

The survey confirmed that the radiological conditions met the NRC guidelines for unrestricted use.

Future Use of Facility

The future use of the facility was not described in the Docket File.

5. License Termination Date

The NRC approved license termination on November 17, 1989.

6. Federal Register Notice

No Federal Register notices are contained in the Docket File.

7. Hazardous Waste Identified at the Site

The Docket File contains no information on hazardous wastes.

8. Restrictions/Conditions/Concerns

The NRC had many initial concerns with the licensee's Decommissioning Plan. However, the licensee responded to the NRC's concerns and adjusted the plan accordingly.

9. Does the Site Meet Present Criteria?

Based on the surveys performed by the licensee and the independent verification contractor, ORAU, the data indicate that the site does meet present criteria.

10. Recommendations

None.

Prepared by: Betty M. Smith
Oak Ridge Associated Universities
Oak Ridge, TN 37830

Date: February 5, 1991

REVIEW OF TEST AND RESEARCH
REACTOR TERMINATED LICENSES

Site:	Memphis State University
Address:	Memphis, Tennessee 38152
Facility:	AGN-201 Research and Training Reactor
Site Contact:	Robert R. Riley, Jr., Safety Services
NRC Contact:	Alexander Adams, Jr., Project Manager, Division of Reactor Projects III, IV, V and Special Projects
Docket No:	50-538
License No:	R-127

1. Summary of Operating History

The model AGN-201 Research and Training Reactor was located in an annex to the ground floor of Building 113 on the MSU campus. The AGN-201 Reactor was first operated by Argonne National Laboratory from 1957-1972. The Reactor had been in storage for about 4 years prior to restoration in 1976 and was operated by MSU until 1985. The AGN-201 Reactor core was never operated at steady state power levels exceeding 0.1 Watt (thermal). Since December, 1976, MSU has conducted more than 2,400 startups and accumulated approximately 960 hours of Operation at Critical.

The AGN-201 reactor consisted of two basic units; the reactor unit and the control console, which was situated in a separate control room. The reactor consisted of the reactor core surrounded by a solid graphite reflector which in turn was enclosed by a lead and water shield. The fuel was distributed within 9 core discs and 4 control rod capsules and was composed of UO2 in polyethylene (approximately 3.3 kg Uranium enriched 19.8% in the isotope U-235). A 10 milligram Radium-Beryllium neutron source and drive assembly was installed in one of the experimental ports of the reactor assembly.

The NRC license R-127 was amended (No. 5) on June 17, 1985 to that of Possession Only status. The sources of radiation (Radium-Beryllium neutron source, reactor fuel (disks and drive rod fuel capsules), were placed in shielded containers or their original shipping container and stored within the facility during dismantling and survey activities.

The Docket File indicates final disposition was made for the component parts and fuel in accordance with the commission's regulations in 10 CFR Chapter I, and in a manner not inimical to the common defense and security, or to the health and safety of the public. The Radium-Beryllium source was transferred to the State of Tennessee under License #R-79177-C-89 and is still maintained in Bldg. 113 of MSU. All fuel was sent to DOE in Oak Ridge, TN. Component Parts were bought by the Institute for Resource Management (IRM) and the Docket File is not specific as to whether IRM has removed the components from the Facility.

2. Request for Termination of License

A Request for Termination of License R-127 was submitted to the NRC on November 10, 1986.

3. Dismantling Order Date

A request for a Dismantling Order and the proposed Dismantling Plan was submitted on November 10, 1986 and approved January 26, 1988.

4. Radiological Status Information

Documentation Identifying Primary Contaminant

The Docket File contains no information identifying a primary residual contaminant. Expected fission product inventory was found to be negligible.

Applicable Guidelines, Supplemental Limits, Exceptions

Regulatory Guide 1.86 was used as applicable guidelines.

Licensee's Survey

A post fuel removal, pre-dismantling survey was performed on March 11, 1985. Very little data is available in the Docket File for this survey. The data that are available are presented in counts per minute with no mention of probe type used.

Measurement, Sampling, and Analyses Procedures

The Dismantling Plan and final report describe all procedures and instrumentation used.

Final Survey Data

A pre-dismantling, post-fuel removal survey was conducted March 14, 1988.

Gamma radiation surveys were conducted using Ludlum Model 19-uR meters. Levels ranged from 4 uR/h to 10 uR/h at various locations in the reactor shield and 6 uR/h to 22 uR/h in the remainder of the reactor building. Background measured was 18 uR/h.

A swipe survey was performed. Swipes were counted using a shielded G-M counting system consisting of a thin window GM tube, a shielded counting pig (Gamma Products Laboratory), and a Nucleus Model 550 Scaler/Timer. The swipes were checked for alpha contamination using a Ludlum model 43-5 Alpha probe in conjunction with a Ludlum Model 3 detector. All swipes were collected over 100 cm^2. Alpha readings for all swipes were < 10 dpm with a BKG of < 10 dpm. Beta-gamma readings ranged from 2609 dpm/100 cm^2 to 5652 dpm/100 cm^2 with a BKG of 4348 dpm/100 cm^2. The highest values reported exceed the 1000 dpm/100 cm^2 removable criteria.

Confirmatory Survey Results

Two separate close-out verification inspections were performed by NRC Region II inspectors. On June 23 and 24, 1988, G. B. Kuzo performed a survey to include review of licensees survey results, radiation exposure surveys, and swipe surveys for removable contamination.

Exposure rate surveys indicated background measurements of 7 uR/h and survey measurements ranging from 3 to 7 uR/h for all areas, excluding Room 147 where the State of Tennessee is maintaining the RaBe source.

Results of swipes of facility and equipment surfaces were reported to be less than the limits specified in Reg. Guide 1.86.

Measurements for fixed alpha and beta were not performed at this time. There is no mention of instrumentation or procedures used to conduct this survey.

On August 4, 1988, supplemental radiological surveys of the facilities and equipment were conducted by a NRC Region II inspector. Alpha and beta-gamma fixed contamination levels were determined for selected facility and equipment surfaces.

NRC survey data indicated less than 52 dpm/100 cm^2 for alpha and less than 2,100 dpm/100cm^2 for beta-gamma.

Instrumentation and procedures used to perform the survey are not described.

Future Use of Facility

The University intends to use the area for other academic purposes.

5. License termination date

   The NRC approved license terminated on October 19, 1988.

6. Federal Register notice.

   Federal Register notices were included in the Docket File.

7. Hazardous Waste Identified at Site

   No hazardous wastes were identified in the Docket File.

8. Restrictions/Conditions/Concerns

   No restrictions, conditions, or concerns were identified in the Docket File.

9. Does Site Meet Present Criteria?

   Probable based on data presented. Several swipes taken by the licensee slightly exceeded removable limits of 1000 dpm/100cm^2, beta-gamma; however, confirmatory measurements suggest it is unlikely removable contamination above guidelines is still present.

10. Recommendations

   Final disposition of the dismantled components should be included in the Docket File. Descriptions of instruments and procedures used by the NRC for confirmatory survey should also be provided, if available.

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Date: February 18, 1991