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AMENDMENTS TO

Application for Renewal of the UCLA Research Reactor License R-71, 1980

Section	Pages to be Removed	Pages to be Inserted (4-30-82)
Front Pages	"General Contents" following transmittal letter (1 page)	Title Page (1 page)
	Table of Contents following "General Contents" (14 pages)	Table of Contents (1 page)
Appendix I	I/1-1 (1 page)	I/1-1,2 (2 pages)
	I/2-1 (1 page)	I/2-1,2 (2 pages)
Appendix II	II/ii (1 page)	II/ii (1 page)
	II/2-1a (1 page)	II/2-1a (1 page)
	II/A-4 (1 page)	II/A-7,8,9,10 (4 pages)
Appendix III	III/i,ii (2 pages)	III/i,ii (2 pages)
	III/1-1,1-3,1-5 (3 pages)	III/1-1,1-3,1-5,1-6,1-7 (5 pages)
	III/6-2,3,4,5 (4 pages)	III/6-2,3,4,5 (4 pages)
	III/8-1 (1 page)	III/8-1 through 8-13 (13 pages)
	III/10-1 (1 page)	III/10-1,2 (2 pages)
	Attachments A & B (16 pages)	Attachments A & B (2 pages)
Appendix IV	ALL (including Attachments A.B.C.D	Reserved New Appendix IV
Appendix V	ALL	New Appendix V

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APPLICATION FOR A CLASS 104 LICENSE

FOR A RESEARCH REACTOR FACILITY

Based on

Code of Federal Regulations, Title 10, Part 50

to

U.S. Nuclear Regulatory Commission

R. R. O'Neill, Dean
School of Engineering and Applied Science
University of California
Los Angeles

February 1980

AMENDED: April 1982

APPLICATION FOR A CLASS 104 LICENSE

FOR A

RESEAERCH REACTOR FACILITY

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4-30-82

FINANCIAL QUALIFICATIONS OF THE APPLICANT

1.0 GENERAL DESCRIPTION

The University of California is a land grant college that is financially supported by:

a) annual appropriations from the State of California;
b) federal and state contracts and grants;
c) student and other user fees; and
d) private gifts and endowments.

There are nine (9) campuses of the University and several laboratories. The "Systemwide Administration" of the University is located in Berkeley, California, adjacent to the University's Berkeley campus. This application pertains to the Nuclear Research Reactor operated by the Nuclear Energy Laboratory (NEL) of the School of Engineering and Applied Science (SEAS) on the University's Los Angeles campus (UCLA).

To the extent that the State Legislature provides funds annually to support the University of California, those funds are distributed to each of the campuses. The funds received by UCLA are further distributed to the various Colleges, Schools and Departments and support activities. Direct support of the NEL derives principally from the operating budget of the SEAS. The SEAS budget is part of the budgeted support for the Los Angeles campus of the University of California. (See UCLA Financial Report, which appears here as Attachment "B"). In addition, the NEL is directly supported by the services of a resident health physicist who is funded out of the budget of the Office of Research and Occupational Safety, an administrative support unit of the campus. UCLA provides indirect support in administrative, custodial, maintenance and surveillance services. Although the actual dollar amount of this indirect support to the NEL cannot be ascertained, an approximation of this amount can be made by applying the rate that is negotiated periodically by the University and the federal government for the recovery of the indirect costs of supporting federal contracts and grants. The current indirect cost rate is used in the total cost analysis that is provided below.

In addition to the direct and indirect support provided by the University through the SEAS at UCLA, the NEL is supported by the recharge income it receives from technical work performed by the NEL staff on contracts and grants of other departments and, to a lesser extent, the fees that are charged for providing reactor services to both academic and non-academic users of the research reactor. The total amount of recharge income and user fee income that is received varies widely from year to year. The initial SEAS budget appropriation is based on an estimate of total expenditures and total income from whatever sources. During financial closing at the end of each fiscal year, the SEAS NEL appropriation is adjusted upwards or downwards to ensure that it equals the total of NEL expenditures less all sources of NEL income.

Funds of the School of Engineering and Applied Science support a broad range of academic programs in furtherance of the University's teaching and research mission. The UCLA Nuclear Energy Laboratory is one such program. Periodically, these programs are subjected to academic review by the faculty of the School. Based on these reviews, recommendations are made to the Dean for continuing financial support. Subject to the availability of funds from the State of California, continuing programmatic need, and continuing positive recommendations by the faculty, the NEL will be maintained at a relatively constant level of financial support adjusted, as needed, for normal increases in costs of operation.

2.0 ESTIMATED ANNUAL COST OF OPERATIONS

The estimated total annual cost of operating the UCLA Research Reactor is the cost of operating the NEL, adjusted to exclude costs associated with the non-reactor-related activities of the laboratory and to include other direct and indirect costs that do not appear in the budget or expenditure statements of the NEL. For the 1980/81 fiscal year these costs are given in the following table which is adapted from the cost accounting data prepared by the UCLA Finance Office. A more complete explanation of NEL operating costs can be obtained from UCLA's letters of January 25 and April 19, 1982 to the Commission.

UCLA Nuclear Energy Laboratory

1980-81 Financial Cost Statement

	Total NEL Budget	Non-Reactor Costs	Net Reactor Costs
Salaries - Permanent Staff of 6 FTE	$163,531	$49,805	$113,726
Salaries - General Assistance	38,265	0	38,265
Employee Benefits	34,288	10,459	23,829
Supplies: $43,406; Equipment: $3,641; Travel: $712	47,759	0	47,759
TOTAL NEL Expense	$283,843	$60,264	$223,579
Additional Expenses not reflected in above totals:			
Health Physicist - Salary	28,000	0	28,000
Health Physicist - Employee Benefits	7,266	0	7,266
TOTAL EXPENSE	$319,109	$60,264	$258,845
Indirect Costs @31% MTDC	97,795	18,682	79,113
TOTAL NEL COSTS	$416,904	$78,946	
TOTAL Reactor Operating Costs			$337,958

Total NEL Expense represents the amount that the NEL had to budget in fiscal year 1980/81 for all its operations. Budget support for the Health Physicist is provided by the Office of Research and Occupational Safety. The precise amount of the indirect costs of reactor operations are unascertainable; however, they are well approximated by the indirect cost rate that has been established for the University as a percentage of modified total direct costs (MTDC), that is, direct costs less equipment. Indirect costs are recovered for the campus as a whole and are not identified in the budgets of individual units such as the NEL. It should be noted that the University's accounting system does nor ordinarily distinguish, within the NEL accounts, reactor-related costs from non-reactor related costs. As one consequence, all of NEL "Salaries - General Assistance" are reported as reactor-related expense. In fact, it is only the salaries of part-time student reactor operators (perhaps $2500 of expense) that is reactor-related. The balance of the part-time salary expense in this category is related to non-reactor projects and activities of the NEL.

In addition to the SEAS appropriated support, the Nuclear Energy Laboratory derives funds by recharging other campus units for technical assistance provided to specific contracts and grants and by charging fees to both academic and non-academic users for reactor services. Support for the Health Physicist (who is budgeted out of the Office of Research and Occupational Safety) and for indirect costs (which are recovered for the campus as a whole) are not considered as sources of funds for NEL operations. The NEL does not regularly issue annual reports of a fiscal nature, however, the approximate distribution of fund sources for the past four (4) fiscal years is shown below.

NEL Sources of Funds

FISCAL YEAR: July 1st to June 30th

	1977-78	1978-79	1979-80	1980-81
SEAS Appropriation	$131,187	$127,636	$151,735	$189,724
Reactor User Fee Income	9,170	11,130	21,000	33,855
Non-Reactor Income	71,675	55,923	67,180	60,264
TOTAL SOURCES OF FUNDS	$212,032	$194,689	$239,915	$283,843

APPENDIX II

ENVIRONMENTAL IMPACT APPRAISAL

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IDENTIFIER	BADGE NUMBER	HIGHEST ANNUAL DOSE	RADIATION	
x^a	219	0 mRem	-----	
x^b	2048	0 mRem	-----	
x^c	1965	1200 mRem	β, γ, x-ray*	[*Note - this badge includes x-ray radiation from the Tokamak Laboratory]
x^d	230	0 mRem	-----	
x^e	1581	0 mRem	-----	
x^f	220	0 mRem	-----	
x^g	1914	0 mRem	-----	
x^h	218	110 mRem	β, γ	
x^i	265	350 mRem	β only	
x^j	203	425 mRem	β only	
x^k	820	0 mRem	-----	
x^l	302	0 mRem	-----	

LOCATION AND MEASUREMENT GUIDE TO FIGURES II/2-1, II/2-2, and II/2-3

UPDATE OF ENVIRONMENTAL MEASUREMENTS

Several developments have occurred subsequent to submittal of the Renewal Application dated February 1980. Firstly, as a result of Question 8 posed by the Nuclear Regulatory Commission on July 31, 1980, UCLA performed a theoretical analysis of plume dispersion based on a Gaussian plume model and showed that such analysis correlated with the previously described dispersion measurements of Rubin. (Analytical results forwarded to the NRC on 9-5-80). Using this dispersion model, the Commission performed calculations of the attendant radiation levels on the roof of the Mathematical Sciences building assuming (conservatively) that the prevailing wind would be realized 100% of the time. These calculations resulted in an estimated dose of 1.4 mRem per year, and hence lead the Commission to respond negatively (on September 24, 1980) to a petition to shutdown the UCLA Research Reactor (Director's Decision under 10 CFR 2.206, DD-80-30).

In addition to these calculations, UCLA initiated a new environmental measurement program utilizing Thermoluminescent Dosimetry (TLD), beginning on August 20, 1980. As a result of what was learned in the 1976-79 monitoring program, dosimeter locations were chosen to minimize the effect of the natural radioactivity of concrete. In general, all dosimeters were placed on non-concrete structures (wood or metal); however, two dosimeters were located in concrete parking structures remote from the reactor to assess radiation levels attributable to concrete. All dosimeters are changed and read quarterly (every three months). Commencing with the second quarter of the study and thereafter, four dosimeters were transferred from raingutters to lead bricks with the bricks interposed between the TLD and the nearest proximate concrete.

The results of the six quarters of TLD observations are shown in Table II/A-1. The geometrical locations of the TLD's specified in that table are graphically illustrated in Figure II/A-3. Starting in the second quarter, lead bricks, 4 x 4 x 2 (inches) were used at locations A, B, D, and E. The bricks were placed on the top surface of the flat roof structure with the TLD fastened to the top of the brick, The brick orientation provided 2 inches of lead shielding between the TLD and the concrete structure. Dosimeters in locations C, G, H, I, J, K, L, and M were fastened to, respectively: the sheet metal of ventilation systems (C, J, M); telescope and planetarium domes (H, K); a wooden housing for meterological equipment (I); and cooling tower windscreens (G, L). TLD F was placed within the exhaust fan inlet plenum chamber and is analagous to TLD No.3 mounted on the stack top in the 1976-79 series.

This monitoring program was initially designed to use thirteen (13) dosimeters at locations A through M. The vendor pricing policy favored using sixteen (16) dosimeters, hence locations O and P were added for the specific purpose of assessing radiation from concrete. Location N

Table II/A-1

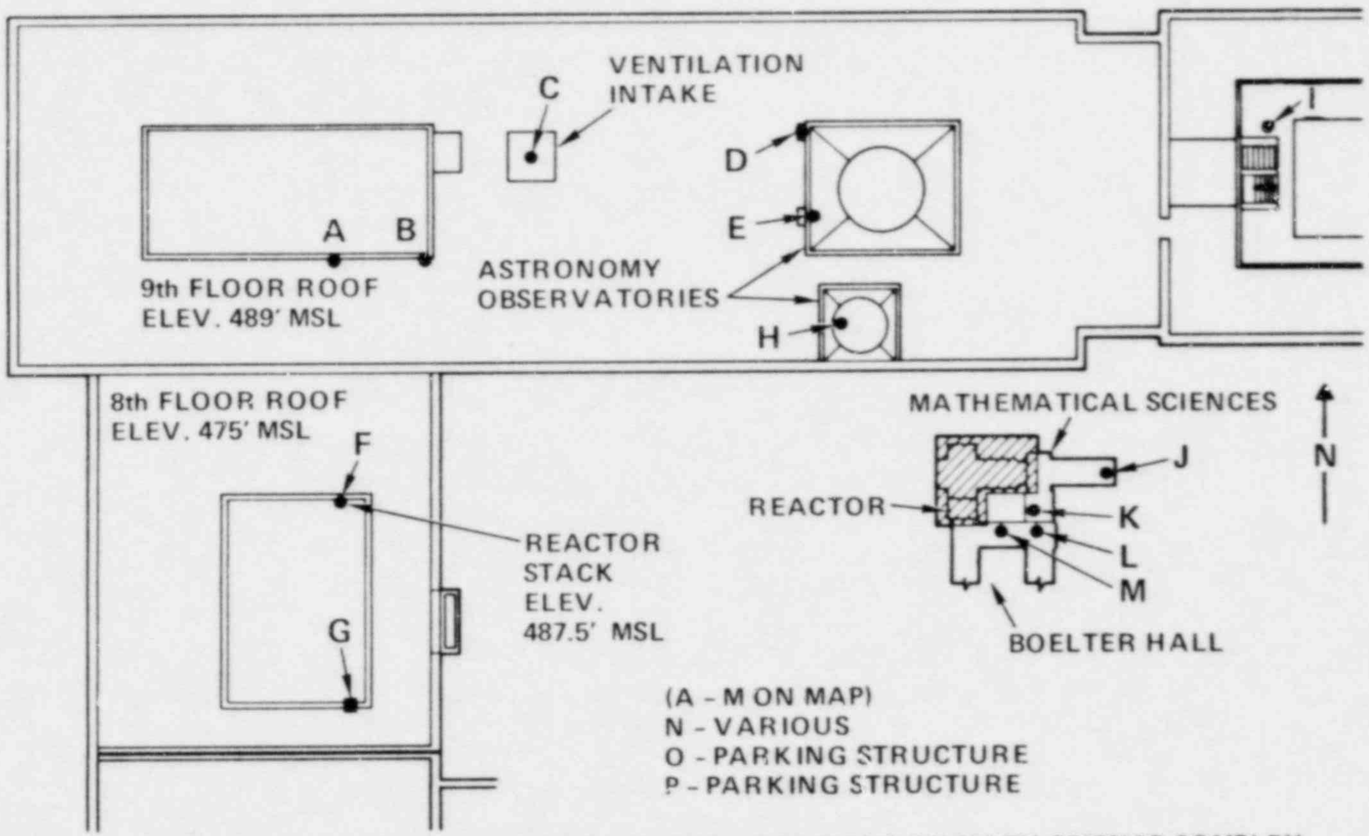
TLD Readings (mRem)

# - LOCATION		8-26-80 12-01-80	12-01-80 3-05-81	3-05-81 5-26-81	5-26-81 8-28-81	8-28-81 11-24-81	11-24-81 2-26-82
A	ROOF TOP, 47'N OF STACK	6	0#	0#	0#	0#	0#
B	ROOF TOP, 50' @ 20°N	5	0#	0#	0#	0#	0#
C	MSA VENTILATION INTAKE, 74' @ 20°N	4	4	4	3	5	5
D	ROOF TOP, 111' @ 51°N	5	0#	0#	0#	0#	0#
E	ROOF TOP, 102' @ 58°N	6	NR*	0#	0#	0#	0#
F	EXHAUST FAN INTAKE PLENUM	12	12	10	14	12	16
G	WINDSCREEN, 38'S OF STACK	3	4	3	2	0	3
H	ROOF TOP, 98' @ 70°N	3	4	4	1	1	0
I	ROOF TOP, 183' @ 68°N	2	5	5	0	3	5
J	ROOF TOP, 353' @ 86°N	0	3	2	0	0	4
K	ROOF TOP, 166' @ 92°N	5	5	4	5	6	3
L	COOLING TOWER, 165' @ 110°N	4	4	3	4	2	3
M	ROOF TOP, 84' @ 148°N	6	6	4	6	6	5
N	VARIOUS	0	LOST	7■	16■	5	5
O	PARKING STRUCTURES	20	16	15	21	17	18
P	PARKING STRUCTURES	21	18	16	15	9	13

#ON LEAD BRICKS

*NOT REPORTED

■DISPLACED FROM ASSIGNED LOCATION



ROOF SAMPLING LOCATIONS ON MATH SCIENCE COMPLEX

Figure II/A-3. TLD Locations

was chosen to replicate an earlier location where the average measured dose of 8.4 mRem per quarter was somewhat intermediate between values typical of concrete-mounted dosimeters and non-concrete mounted dosimeters. It was the only dosimeter mounted on a lead brick in the first quarter. The value for the first quarter was very low, but successive thefts of the lead bricks in the second and third quarters discouraged the continued use of that location. Therefore, dosimeter N was relocated on a wooden tower; however, it was somehow displaced during the quarter and the reading for the fourth quarter was compromised. Although this badge remained on the tower during the fifth quarter, a decision was made to move the dosimeter to an entirely different location. For the sixth (and current) quarter, the dosimeter has been mounted on the windscreen surrounding the stack. The location is symmetrical relative to concrete walls and parapets, and relative to the TLD in the exhaust fan intake plenum. The objective has been to distinguish between an immersion dose and a background dose in otherwise similar locations.

TLDs O and P were placed in parking structures north of the reactor building for the first three quarters and then placed in parking structures generally west of the reactor for the next three quarters. The location change was made to broaden the sample base.

The radiation levels seen by the TLDs in parking structures (12 readings) averaged 66 mRem per year whereas the TLD in the exhaust fan intake plenum averaged 51 mRem per year. The conclusion that concrete is a source of radiation is inescapable, but the quantitative contribution of this radiation source to arbitrarily placed TLDs is not readily estimated. The TLDs placed on lead bricks showed zero or slightly negative background values even though these locations were in the general downwind direction of the plume. The zero or negative background values are to be expected in that the lead bricks shield out the normal terrestrial component of the natural background radiation, and the reactor exhaust plume contributes no measurable increase in the background downwind from the stack. The average value of all other dosimeters (8 in number, 48 observations) in the roof top vicinity of the stack is 13.6 mRem per year.

The results of this second TLD program indicate that radiation from the plume is low, but that individual observations are probably sensitive to geometry, proximity of concrete, and shielding. A complete separation of the low level plume radiation from natural and man-enhanced (concrete) radiations does not appear to be feasible using TLDs.

APPENDIX III

ARGONAUT SAFETY ANALYSIS REPORT (ASAR)

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APPENDIX III

ARGONAUT SAFETY ANALYSIS REPORT (ASAR)

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ARGONAUT SAFETY ANALYSIS REPORT (ASAR)

1.0 INTRODUCTION AND GENERAL DESCRIPTION

1.1 Description

The Argonaut Safety Analysis Report has been prepared for submission to the U.S. Nuclear Regulatory Commission in support of reapplication for an Operating License. The application is made by the Regents of the University of California for the continued operation of the reactor, licensed as R-71, at the Los Angeles Campus.

The plant housing the reactor is located in the northwest wing of Boelter Hall at the University of California, Los Angeles. The 400-acre campus is located on a coastal plain and is approximately five miles east of the Pacific Ocean and fifteen miles west of the Los Angeles civic center. To the south of the campus is a business and shopping district, and to the north, west and east are residential areas. A map of the general area is shown in Figure III/1-1.

The reactor is located at the Nuclear Energy Laboratory in a 2 story, reinforced concrete structure with a floor area of approximately 75 x 90 ft. and a height of 27 feet. The construction of the reactor facility began in 1959, with the assistance of a $203,350 grant from the U.S. Atomic Energy Commission, through the efforts of the founding Director, Dr. Thomas E. Hicks. This grant was disbursed in construction and reactor equipment.

Subsequent construction, completed in 1968, has surrounded the reactor room on the north, east and south sides with additional laboratory space that provides a buffer zone between the reactor room and adjacent but unrelated facilities. On the west side, first floor laboratory spaces and the second floor control room intervene between the reactor room and the exterior building wall.

The third floor (roof) of the reactor building is bridged by new construction at the fifth, sixth, seventh, and eighth (roof) levels. The void region between the third and fifth floors is a limited access region which contains a small structure housing air conditioning and water demineralization equipment.

The nuclear reactor is an Argonaut type; water-cooled and moderated, graphite reflected, 93% enriched uranium thermal reactor, that is currently licensed for a maximum core thermal power of 100 kw. By special amendment, the reactor has operated in the past for brief periods of up to 500 kw. It appears that the reactor could safely operate up to 1,000 kw with modifications to the shielding, the cooling system, and special provisions for reducing argon-41 emission.

1.2 Operations

Historically, the UCLA reactor reached criticality on October 21, 1960, at 6:54 p.m. After a program of low-power testing at 10 watts, the reactor went to its then licensed power of 10 kw in February of 1961. The reactor was modified slightly, license amendments were approved, and in October of 1963, the reactor reached its present licensed full thermal power output of 100 kw. The chronology of these and other events is shown in Table III/1-1.

The reactor generates no electricity and is used primarily for activation analysis, class instruction, student experiments and faculty, staff, and student research. To provide this flexibility, the reactor has three vertical irradiation holes (1.9" ID), a 78 cubic foot removable graphite thermal column with a one cubic foot irradiation volume, two 6" ID and four 4" ID horizontal beam ports, and a 3,000 gallon water-filled irradiation volume. A pneumatic transfer system ("rabbit") provides sample irradiation in the west vertical port with rapid transfer to a counting laboratory.

The variety of irradiation ports has provided great flexibility in the kinds of experiments that can be conducted with the reactor. The fast and thermal flux is maximized in the vertical ports, the thermal to fast flux ratio is maximized in the thermal column and a neutron and/or γ beam may be extracted from the horizontal beam ports. Table III/1-2 gives a brief description of the annual reactor use from 1973 to 1981. Variations from year to year are attributed to research demand, changes in technology, random maintenance requirements, class scheduling, and enrollments.

Class instruction includes the instruction of undergraduate and graduate students of the UCLA School of Engineering and Applied Sciences and other departments in basic nuclear engineering theory and applications. Class instruction also includes general health physics and reactor operator training. Table III/1-3 lists the current class offerings which require use of the research reactor and the total annual student hours of reactor dependent instruction projected for the 1981/82 academic year.

When not being used for class instruction, the reactor is made available to assist both academic and non-academic users in activation analysis, delayed-neutron counting, fission track dating projects, and other experimental techniques. All such non-instructional uses of the reactor have been categorized as research. A number of the academic users of the facility are from other colleges and universities in the area. Recently, a non-academic user of the reactor has been employing activation analyses techniques in his ore-assaying business. All research users of the facility are charged a fee for the reactor services provided. The fee is based on "port-hours" of reactor operation. Although up to four (4) experimental ports may be used during one hour of actual reactor operation, such use is rare because of demand and incompatibility of desired irradiation conditions. The port-hours of use by each category of research user during the past ten calendar years is given in Table III/1-4.

Table III/1-2

REACTOR ANNUAL USE			
Year	Number of Runs	Megawatt-Hours	Actual Operating Hours
1973	76	13.8	
1974	76	14.8	
1975	91	11.9	
1976	82	13.1	184
1977	106	15.9	238
1978	132	20.3	271
1979	149	29.0	372
1980	131	28.9	381
1981	134	23.9	364

Table III/1-3

UCLA NUCLEAR ENERGY LABORATORY

Table of Class Use of UCLA Reactor 1981 - 1982 Academic Year

USE → / ↓ CLASS	UNITS PER QUARTER	STUDENTS PER PER QUARTER	REACTOR ACADEMIC HRS/QTR [2]	LABORATORY ANALYSIS HRS/QTR [3]	LECTURE & PREPARATION HRS/QTR [4]	TOTAL HRS/QTR	STUDENT HRS/QTR	OFFERINGS PER YEAR	STUDENT HOURS/YEAR
ENGR 135 Al	2	8	9	2	29	40	320	1	320
ENGR 135 BL	2	8	9	4	27	40	320	1	320
ENGR 135 F	2	5	28 (100)[5]	0	12	40	200	1	200
ENGR 139 A	4	25	1	12	7	20	500	3	1500
CHEM 184 A	4	16	1	7	2	10	160	1[6]	160
E&SS 298	4	6	1	32	15	48	288	1	288
PHYS 180 A	4	10	1	12	11	24	240	1[6]	240
ENGR-EXT. 497.17	4	10	3	0	27	30	300	1	300
TOTAL:	ANNUAL STUDENT HOURS OF REACTOR DEPENDENT INSTRUCTION								3328

[1] CLASSES LISTED ARE THOSE WHICH USE THE REACTOR FOR THE INSTRUCTION OF UCLA STUDENTS IN THE SCHOOL OF ENGINEERING, AND THE DEPARTMENTS OF CHEMISTRY, EARTH AND SPACE SCIENCE, AND PHYSICS IN REACTOR CHARACTERISTICS, BOTH FUNDAMENTAL AND OPERATIONAL, ACTIVATION ANALYSIS, AND REACTOR OPERATIONS. THE TABLE DOES NOT INCLUDE CLASSES FROM OTHER COLLEGES AND UNIVERSITIES WHICH USE THE REACTOR. STUDENT ENROLLMENT IN THESE COURSES AND THE SPECIFIC COURSE CONTENT VARIES FROM ACADEMIC QUARTER TO ACADEMIC QUARTER. THE TABULATED ENTRIES REPRESENT THE CURRENT TYPICAL USAGE AS ESTIMATED BY THE COURSE INSTRUCTORS.

[2] REACTOR ACADEMIC HOURS - INCLUDES OPERATING HOURS "AT-POWER" AS REPORTED ANNUALLY TO THE NRC AS WELL AS "NON-POWER" HOURS SUCH AS THE "APPROACH-TO-CRITICAL" EXPERIMENT IN ENGR 135 AL AND THE PRE-START CHECK-OFF IN THE OPERATOR TRAINING COURSE ENGR 135 F.

[3] LABORATORY ANALYSIS HOURS - RECOGNIZES THE USE OF THE REACTOR IN THE PRODUCTION OF VARIOUS RADIOACTIVE MATERIALS OR SUBSTANCES WHICH SUBSEQUENTLY ARE SUBJECTED TO LABORATORY ANALYSIS BY STUDENTS, FOR EXAMPLE, TO PRODUCE MATERIALS USED IN GAMMA RAY SPECTROSCOPY.

[4] LABORATORY LECTURE AND PREPARATION HOURS - RECOGNIZES THE STUDENT INSTRUCTION THAT OCCURS IN CONNECTION WITH THE OPERATION OF THE REACTOR IN REACTOR PHYSICS AND OPERATIONS, REACTOR INSTRUMENTATION, EXPERIMENTAL PROCEDURES AND TECHNIQUES, MEASUREMENT TECHNIQUES, AND METHODS OF DATA REDUCTION.

[5] INCLUDES APPROXIMATELY 100 ADDITIONAL TRAINING HOURS REQUIRED FOR OPERATOR LICENSING, THE TRAINING TAKING PLACE CONCURRENTLY WITH OTHER REACTOR OPERATIONS.

[6] GENERALLY TWO COURSES WITH DIFFERENT COURSE CONTENT BUT WITH THE SAME COURSE NUMBER ARE OFFERED ANNUALLY, ONLY ONE OF WHICH REQUIRES THE USE OF THE REACTOR.

Table III/1-4

Research Usage of the Reactor

User Category	Port Hours										Total Port Hours
	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	
NEL Staff Users	41	1	31	11	4	31	9	1	27	113	269
Other UCLA Users	81	122	105	139	109	106	105	91	101	67	1026
College & Univ. Users	25	31	45	27	45	47	37	53	20	38	368
Non-academic Users	2	1	--	1	1	5	95	264	360	211	940
Total Port Hours	149	155	181	178	159	189	246	409	508	429	2603

TABLE III/6-1(a) COMPARISON TABLE - GENERAL

REACTOR LOCATION	UNIVERSITY OF CALIFORNIA, LOS ANGELES LOS ANGELES, CALIFORNIA 90024	UNIVERSITY OF FLORIDA GAINESVILLE, FLORIDA, 32611	UNIVERSITY OF WASHINGTON SEATTLE, WASHINGTON, 98195
LICENSE NO.	R-71	R-56	R-73
DOCKET NO.	50-142	50-83	50-139
OWNED BY	REGENTS OF THE UNIVERSITY OF CALIFORNIA	COLLEGE OF ENGINEERING	UNIVERSITY OF WASHINGTON
OPERATED BY	NUCLEAR ENERGY LABORATORY, UCLA	DEPARTMENT OF NUCLEAR ENGR. SCIENCE, U. OF FLORIDA	NUCLEAR ENERGY DEPARTMENT COLLEGE OF ENGINEERING, U OF W
REACTOR TYPE	HETEROGENEOUS, THERMAL, LIGHT WATER COOLED & MODERATED, GRAPHITE REFLECTED, 93% ENRICHED URANIUM	*, EXCEPT ORIGINAL FUEL WAS 20% ENRICHED	*
DESIGNED BY	GENERAL NUCLEAR ENGR. (PRINCIPAL CONSULTANT TO REACTOR SYSTEM) STANTON & STOCKWELL, ARCHITECT AMF (REACTOR)	GENERAL NUCLEAR CORP., (REACTOR SYSTEM) G.C. FULTON, ARCHITECT TO THE STATE BOARD OF CONTROL (BUILDING) *	LOVETT, STREISSGUTH & ZEMA ARCHITECT *
CONSTRUCTION BY	JONES BROTHERS CONST. CO. & LOUIS C. DUNN INC.-BUILDING A.I.-FUEL ELEMENTS (2ND TIME) HONEYWELL-CONTROLS, ELECTRONICS	COOPER CONSTRUCTION CO.-BUILDING AMF ATOMICS-REACTOR COMPONENTS SYLCOR NUCLEAR-FUEL ELEMENTS HONEYWELL-CONTROLS	JENTOFT & FORBES-CONTRACTOR THE MARTIN CO. (FUEL SET)
ORIGINAL DESIGNED POWER	100KW	*	*
NORMAL OPERATING POWER	100KW (1963)	* (1966)	* (1967)
NORMAL POWER (LINEAR, SPECIFIC POWER)	.19KW/FT, 28.12KW/KG U^{235}	*	*
OPERATING SCHEDULE	VARIABLE, 13-16 MEGAWATT HRS PER YEAR	VARIABLE, 15 TO 20 MW HRS PER YEAR	VARIABLE
PRINCIPAL USE OF REACTOR	ACTIVATION ANALYSIS, REACTOR OPERATOR TRAINING, AND EDUCATION OF NUCLEAR ENGR. & STUDENTS	TRAINING AND EDUCATION OF NUCLEAR ENGR. AND SCIENTISTS, REACTOR OPERATORS, ACTIVATION ANALYSTS & LASER EXP'S.	REACTOR OPERATOR TRAINING, CLASS DEMONSTRATIONS, ISOTOPE PRODUCTION, INAA RESEARCH
OPERATING STAFF	3 PER SHIFT, 1 SHIFT 4 PART TIME STUDENT OPERATORS	2 PER SHIFT, 1 SHIFT NORMALLY 2 PART TIME OPERATORS	2 PER SHIFT, 1 SHIFT
STATUS: DATE FIRST CRITICAL	10/60	1/59	4/61
DATE FULL POWER 10KW, 100KW	2/61 & 10/63	4/59, 8/62	4/61, 5/67
SOLID FUEL: FUEL ELEMENT A) SHAPE	FLAT PLATE (MTR TYPE)	*	*
B) FUEL COMPOSITION	13.4 WT.% U-AL ALLOY	*	*
C) FUEL DIMENSIONS (MEAT)	24" x 2" x .040"	*	23.5" x 2.3" x .040"
D) CLADDING MATERIAL	1100 ALUMINUM	*	*
E) CLADDING THICKNESS	.015 IN.	*	*
F) TYPE OF SUBASSEMBLY	PARALLEL PLATES BOLTED TOGETHER	*	*
G) # OF ELEMENTS PER ASSEMBLY	11	*	*
H) PLATE DIMENSIONS	25-5/8 x 2-7/8 x .070 IN.	*	*
I) SUB-ASSEMBLY (FHU) DIMENSIONS	27" (INCLUDES HANDLING RING) x 2-7/8" x 2-3/8"	*	*
J) # OF FHU'S	24	*	*
K) ARRANGEMENT OF SUB-ASSEMBLY	4 ELEMENTS IN EACH OF 6 BOXES ARRANGED IN 2 PARALLEL ROWS	*	*
L) LIFETIME	INDEFINITE	*	*
METHOD OF REFUELING	MANUAL, USING HAND TOOLS, TRANSFER CASK, REMOTE MIRROR	*	*

TABLE III/6-1(b) PRIMARY COOLANT

FLUID	H_2O	•	•
CIRCULATION A) DIRECTION OF FLOW	UPWARDS	•	•
B) FLOW INDUCED BY	FORCED FEED, GRAVITY RETURN	•	•
C) NORMAL FLOW RATE	16 GPM	40 GPM	23 GPM
D) MEAN VEL. THRU CORE	1.3 cm/SEC	4 cm/SEC	2.1 cm/SEC
E) AVERAGE INLET TEMP.	100°F	96°F	98°F
F) CORE ΔT	42°F	16.6°F	28°F
HEAT DISSIPATION METHOD	WATER-TO-WATER TUBE-SHELL S.S. HEAT EXCHANGER DOUBLE-PASS SECONDARY (TUBE)	•-U TUBE PRIMARY, SINGLE PASS SECONDARY-(1 MW RATED)	•
MEANS OF PURIFICATION	PRIMARY SYSTEM: CONTINUOUS ½ GPM BYPASS THRU CARTRIDGE FILTER THEN THRU DEMINERALIZERS AND RETURN TO DUMP TANK SHIELD TANK: INDEPENDENT, CONTINUOUS, 30 GPM, FILTER AND 2 PARALLEL DEMINERALIZER CARTRIDGES	•	•(900 ML/MIN)

NUCLEAR DATA

FUEL LOADING A) COLD CLEAN CRITICAL MASS	3.2 KG U^{235}	•	3.29 KG U^{235}
B) NORMAL FRESH FUEL LOADING	3.6 KG U^{235}	•	3.439 KG U^{235}
C) EXCESS K, FRESH LOADING	2.3% Δk/k	•	•
FLUX @ 100KW A) PEAK THERMAL FLUX	1.5×10^{12} n/cm² sec		1.4×10^{12} n/cm² sec
B) PEAK FAST FLUX	1.5×10^{10} n/cm² sec	•	5.0×10^{11} n/cm² sec
C) PEAK THERMAL COLUMN THERMAL FLUX	1.0×10^{10} n/cm² sec		3.0×10^{10} n/cm² sec
REACTIVITY COEFFICIENTS (WATER) A) TEMP.	-0.48×10^{-2}% Δk/k (-0.74¢/°F)	-1.0×10^{-4} Δk/k/°C	-0.00F Δk/k/°F (H_2O), + 0.0014% Δk/k/°F (GRAPHITE)
B) VOID	-0.20 Δk/k% H_2O VOID (-25¢/% VOID)	-0.21 Δk/k/% H_2O VOID	.0002% Δk/k/cm³ VOID
C) MASS COEFFICIENTS U^{235}	0.31% Δk/k% U-235	•	0.01% Δk/k/GM U^{235}
D) CORE EXCESS	2.3% Δk/k \$3.54	•	
BURNABLE POISONS	NONE	•	•
NEUTRON SOURCE	6.6 mCi RaBe	UP TO 5 Ci Sb-Be AND 1 Cu PuBe	Sb-Be (2 Ci PuBe IF NEEDED)

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TABLE III/6-1(c) REACTOR CHARACTERISTICS

ℓ_p	1.4×10^{-4} SEC		
ASSUMED ℓ_p	2×10^{-4} SEC		2.55×10^{-4} SEC (MEASURED)
$\bar{\tau}$	.085 SEC		
ASSUMED $\bar{\tau}$	.085 SEC		
β^*	~.0070		
ASSUMED β_{EFF}	.0065		.0074
BIOLOGICAL SHIELD	CONCRETE	CONCRETE	CONCRETE
FULL SCRAM (DROP ROD & WATER)	A) POWER FAILURE B) MANUAL (FULL SCRAM BUTTON) C) SHORT PERIOD (LESS THAN 3 SEC) D) HIGH FLUX (POWER > 125 KW) E) CLOSURES OPEN ABOVE 1 WATT F) DUMP VALVE OPEN	A) MANUAL SCRAM BAR B) 3 SECOND OR LONGER C) 1 OF 2 125% (125 KW) (2 SAFETY CHANNEL) D) NONE E) RESULTS IN LOW PC LEVEL, AND NO PC FLOW SCRAM. INTERLOCKS WITH PC PUMP. KEY TURNED OFF WITH 2 OR MORE BLADES UP = FULL SCRAM WITH WATER DUMP	A) POWER FAULURE B) MANUAL SCRAM C) SHORT PERIOD < 3 SEC D) HIGH FLUX = 125 KW E) OPEN DUMP VALVE
DROP ROD SCRAM	A) KEY TURNED OFF B) LOSS OF HIGH BAY VENTILATION C) LOSS OF PRIMARY PUMP POWER D) LOSS OF PRIMARY COOLANT FLOW E) LOW CORE WATER LEVEL F) LOW SHIELD TANK WATER LEVEL G) HIGH SECONDARY EFFLUENT RADIATION MONITOR (> 8×10^{-6} µCI/ML I-132 IN H_2O) H) DROP ROD BUTTON I) LOGIC CONDITION	A) IF ONLY 1 BLADE UP = MAG CLUTCHES OFF B) LOSS OF DILUTING OR VENT FAN POWER C) LOSS OF PC PUMP POWER D) PC FLOW DROPS TO 30 GPM FROM 40 GPM E) PC WATER LEVEL DROPS ABOUT 2 INCHES (STILL ABOVE FUEL PLATES) F) SMALL DROP - MOVABLE LEVEL SWITCH G) NO SCRAM. WEEKLY SAMPLE ANALYZED	A) TURN KEY OFF B) LOW FLOW PRIMARY C) LOW CORE LEVEL PRIMARY D) LOSS PRIMARY FLOW E) LOW SHIELD TANK WATER LEVEL F) DROP ROD BUTTON G) DILUTION FAN (AIR EXHAUST)
INHIBITS	A) NEUTRON START-UP SOURCE COUNT LESS THAN 2 CPS B) PERIOD LESS THAN 6 SEC C) CLOSURE OPEN (BELOW 1 W) D) LOG-N AMPLIFIER NOT IN OPERATE MODE	A) SAME AS SCRAM BAR B) LOSS OF 2° WATER ABOVE 1 KW. PC TEMP 155°F OR HIGHER C) 10% DROP IN NEUTRON CHAMBER HIGH VOLTAGE, EITHER OF 2 HV SUPPLIES D) 2 CPS INHIBIT OF UP DRIVE ONLY E) LOGIC - CANNOT WITHDRAW 2 OR MORE BLADES SIMULTANEOUSLY F) 10 SECOND PERIOD UP DRIVE G) NONE H) WIDE RANGE (LOG N + STARTUP) CHANNEL ALWAYS IN OPERATION. INHIBIT IF CALIBRATE OR TRIP TEST SWITCHES NOT IN OPERATE MODE (FOR UP DRIVE)	A) PERIOD < 10 SEC B) CORE INLET < 85°F C) BF_3 < 10 CPS/SEC
ALARMS (LIGHT & HORN)	A) HIGH PRIMARY COOLANT EXIT TEMPERATURE - 180°F 2) HIGH AREA RADIATION > 5 MR/HR NORTH & SOUTH HIGH BAY > 10 MR/HR RADIOACTIVE STORAGE > 100 MR/HR RABBIT ROOM C) ARGON 41 IN STACK > $1.8 \times 10^{-?}$ µCI/ML IN AIR	A) 150°F = AUDIBLE ALARM B) ANY OF 3 AREA MONITORS = ALARM LIGHT @ 2.5 MR/HR = AUDIBLE ALARM & LIGHT @ 10 MR/HR. TWO CHANNELS @ 10 MR/HR ALSO = EVACUATION ALARM SIRENS. LOSS OF GREEN "NO FAIL" LIGHT IF CHANNEL MODULE NOT RECEIVING SIGNAL FROM AREA MONITOR CHAMBER C) 2 CUPS WATER IN PC PIT SUMP = AUDIBLE ALARM IN CONTROL ROOM, LIGHT D) ALARM LEVEL, STACK EFFLUENT ADJUSTABLE ACCORDING TO POWER OF OPERATION	A) HIGH PRIMARY EXIT SET 140°F B) HIGH RADIATION AREAS REACTOR FLOOR > 1.5 MR/HR RABBIT EXIT > 10 MR/HR REACTOR TOP > 5 MR/HR HIGH ARGON 41 C) ANY SCRAM OR ROD DROP
ALARM LIGHT ONLY	A) LOW PRIMARY COOLANT RESISTIVITY < 1×10^{6} OHMS B) BF_3 DETECTOR IN HIGH FLUX > 0.022 WATTS C) ANY INHIBIT	A) LOSS OF POWER OR OPEN CIRCUIT IN EVACUATION SIREN = LOSS OF GREEN LIGHT B) PC SOLU BRIDGE MONITORS D/M INLET AND OUTLET RESISTIVITY ALARM LIGHT ADJUSTABLE ON INSTRUMENT C) AIR CONDITIONER TRIP BY SIREN = RED LIGHT D) B PROPORTIONAL COUNTER WHEN ENERGIZED HAS RED "EXTENDED RANGE" LIGHT. AUTOMATIC HV CUT OFF @ 400 CPS E) FAST PERIOD INHIBIT LIGHT & SWITCHES F) FAST POWER INHIBIT LIGHT & SWITCHES NOT IN OPERATE INHIBIT MODE TWO ALARM LIGHTS FOR 2° WATER FLOW (140 GPM & 60 GPM) 60 GPM ALARM = SCRAM AT OR ABOVE 1% POWER AFTER 10 SECOND WARNING LIGHT B) LOSS OF LIGHTS = FAILURES IN RAD MON STANDBY BATTERY POWER PACK C) AUDIBLE ALARM & RED LIGHT IF BACK DOOR IS OPENED. RED AND GREEN LIGHTS MONITOR OUTER ACCESS DOORS TO FACILITY	A) LOW PRIMARY CONDUCTIVITY < 500,000 OHM B) BF_3 HIGH FLUX > 2 WATTS C) INHIBITS

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TABLE III/6-2 TRAINING REACTOR CHARACTERISTICS

DATE	1960	1980
TYPE	HETEROGENEOUS, THERMAL	•
POWER	10 KW	100 KW
FLUX LEVEL (AT 10 KW)	1×10^{11} N/CM^{2}SEC	1.5×10^{12} N/CM^{2}SEC
EXCESS REACTIVITY (TECH SPEC LIMIT)	0.58% ρ AT 32°F	2.3% ρ AT ROOM TEMP
EXCESS REACTIVITY INSTALLED	1.5% ρ AT ROOM TEMP	1.8% ρ AT ROOM TEMP
CLEAN COLD CRITICAL MASS	3200 GM U-235	•
EFFECTIVE PROMPT NEUTRON LIFETIME	1.4×10^{-4} SEC	2×10^{-4} SEC
UNIFORM WATER VOID COEFFICIENT	-0.19% ρ/% VOID	-0.164% ρ/% VOID
TEMPERATURE COEFFICIENT	-0.48×10^{-2}% ρ/°F	$-.481 \times 10^{-2}$% ρ
U-235 MASS COEFFICIENT	+0.31% ρ/% U-235 MASS	+.3% ρ/% U-235
START-UP SOURCE	2 CURIE Pu Be	6.6 mCi Ra Be
REFLECTORS	GRAPHITE (1.67 GM/CC)	•
MODERATOR	H_2O AND GRAPHITE	•
DELAYED NEUTRON FRACTION	0.0068	.0065
FUEL PLATES		
FUEL	93% ENRICHED, U-AL ALLOY	•
FUEL LOADING	3,445.2 GM U-235	3,556 GM U-235
PLATE THICKNESS	0.070 IN.	•
WATER CHANNEL	0.137 IN.	•
ALUMINUM TO WATER RATIO (VOL)	0.51	•
MEAT COMPOSITION	13.4 WT% U-AL	•
COOLANT	H_2O	•
FLOW	10 GPM	16 GPM
TEMPERATURE, IN	103°F	100°F
TEMPERATURE, OUT	110°F	142°F
CONTROL BLADES	Cd, SWINGING VANE, GRAVITY FALL	•
NUMBER	3 SAFETY: 1 REGULATING	•
INSERTION TIME	0.324 SEC (CALCULATED)	0.5 SEC (MEASURED)
REMOVAL TIME	90 SEC (MINIMUM)	100 SEC
BLADE WORTH, SAFETY	3 RODS 1.5% ρ = 4.5% ρ	3 RODS ~ 1.6% ρ = 4.87
BLADE WORTH, REGULATING	1 ROD 0.6% ρ = 0.6% ρ	1 ROD ~ 1% ρ
	TOTAL = 5.1% ρ	TOTAL ~ 5.8% ρ
REACTIVITY ADDITION RATE, MAX.	0.02% ρ/SEC	.05% ρ/SEC
SHIELD (CONCRETE)		
SIDES, CENTER	6 FT. 0 IN. CAST, MAGNETITE	•
SIDES, SHIELD TANK END	6 FT. 8 IN. CAST, ORDINARY	•
SIDES, THERMAL COLUMN END	6 FT. 8 IN. CAST, MAGNETITE	•
MIDDLE	CAST CONCRETE BLOCKS	•
ABOVE CORE	5 FT. 10 IN. MAGNETITE BLOCKS	• PLUS 39" OF BORATED PARRAFIN
ENDS	3 FT. 4 IN. MAGNETITE BLOCKS	•
EXPERIMENTAL FACILITIES		
THERMAL COLUMN, HORIZONTAL	5 FT. x 5 FT. x 4 FT. 11 IN. LONG	60 IN. x 52 IN. x 43 IN. LONG REMOVABLE
THERMAL COLUMN, VERTICAL	PROVISION FOR INSTALLATION	•
SHIELD TEST TANK	5 FT. x 5 FT. x 14 FT. 6 IN. HIGH	•
EXPERIMENTAL HOLES	2 - HORIZONTAL, 6 IN. DIAMETER	•
	5 - HORIZONTAL, 4 IN. DIAMETER	4 - HORIZONTAL, 4 IN. DIAMETER
	3 VERTICAL, 1½ IN. DIAMETER	3 - VERTICAL, 1-7/8 IN. DIAMETER
EXPERIMENTAL HOLES, THERM. COL.	15 REMOVABLE GRAPHITE STRINGERS	•
FOIL SLOTS	11 - HORIZONTAL, 1/8 IN. x 1/2 IN.	•
	16 - VERTICAL, 3/8 IN. x 1 IN.	•

8.0 CREDIBLE ACCIDENTS FOR ARGONAUT REACTORS

8.1 INTRODUCTION

The original accident analysis for the UCLA Argonaut-Type reactor, which was completed in 1960, postulated an accident involving local melting of the fuel core and assumed that a release of 10% of the volatile fission products into and away from the reactor building would result [1]. The authors of that work provided no basis for the postulate, but noting the inherent self-limiting characteristics of the reactor they did state that they regarded any core melting as not plausible. The UCLA reactor technical staff has always considered accidental core-melting to be implausible. With the recent release of certain generic analyses by the NRC, the implausibility of a core-melting accident has been confirmed for all Argonaut-type research reactors.

The postulatory basis for the core melt scenario discussed in the original UCLA accident analysis was probably related to the concept of a "Maximum Credible Accident" for power reactors. In power reactors, core-melting damage can be causally related to inadequate heat removal following a loss of coolant accident. In Argonaut-Type research reactors, decay heat power density is far less, and loss of coolant is a designed back-up shutdown or "scram" system - a safety feature rather than a hazard.

The NRC's generic analyses, which are discussed below, have served to identify the self-limiting characteristics of Argonaut reactors. These generic studies, which in all cases were based upon very conservative assumptions and analyses, demonstrate that accidental core melting of a 100 kw Argonaut reactor (such as the UCLA research and teaching reactor) is a non-credible event. Thus, for an Argonaut reactor, there is no equivalent to the "Maximum Credible Accident" of power reactors and because that phrase carried the connotation of core melting it is not used further in this analysis. The general conclusion of the generic studies is that credible accidents hypothesized for Argonaut reactors predict relatively minor radiological consequences. This conclusion is fully applicable to the specific case of the UCLA research reactor. Among the accidents examined in the generic studies, a fuel-handling accident was determined to be the worst credible accident and has therefore been adopted by UCLA as the design basis for emergency response planning, and is discussed in detail below.

8.2 GENERIC STUDIES

Two generic accident analyses of Argonaut reactors have been released: one by Battelle Pacific Northwest Laboratory [8]; and one by Los Alamos National Laboratory [9]. These two studies have been incorporated herein by reference as Attachments "A" and "B", respectively, to this Appendix. Drawing on these studies, the NRC has produced its Safety Evaluation Review (SER) of the UCLA

8.3 GENERIC STUDIES EXTENDED FOR THE CASE OF THE UCLA REACTOR

The Argonaut generic studies and other matters particular to the UCLA facility have been reviewed by the NRC. In its SER for the facility, the NRC noted that extremely conservative assumptions and analyses were used in the Battelle and Los Alamos studies. As a result unduly conservative estimates were made of the predicted consequences of either a fuel-handling or earthquake-core-crushing accident. Therefore, the SER treated the fission product release that might result from a fuel-handling accident as calculated in the Battelle study as equivalent to the fission product release that might result from a severly damaged fuel core caused by a building collapse during a major earthquake.

The conservative analyses of the Battelle and Los Alamos studies served to strongly support the NRC's general conclusion that no significant radiation hazards to individuals in either restricted or unrestricted areas would result from accidents at the UCLA reactor. However, in examining the consequences of credible accidents at the UCLA facility for the purpose of planning emergency responses, it is necessary to extend the generic studies to take into account certain site-specific factors. In the discussion which follows, it will be useful to distinguish radiological accidents that might occur as a secondary result of some naturally occurring event, such as an earthquake, and accidents that might occur during the ordinary course of reactor or facility operations. It may be assumed, with respect to accidents that occur as a secondary result of some natural event, that any additional hazard that might be hypothesized due to the existence of the reactor would be inconsequential relative to the general disaster caused by such an event.

8.3.1 CATASTROPHIC SEISMIC EVENT

Due to the fact that the predicted consequences of a crushed reactor core are relatively insignificant [8,9], a detailed seismic analysis of the UCLA facility is not warranted. The following remarks are only intended to suggest certain of the factors that would be relevant in the prediction of the radiological consequences of a seismic event at the UCLA facility.

The known geological faults closest to the UCLA campus are the Newport-Inglewood fault to the east and the Santa Monica-Hollywood fault to the South [12,13]. Since the Newport-Inglewood fault is estimated to be capable of generating an earthquake of magnitude 7 to 7.5 (Richter) with a recurrence period of 1000 years, it is regarded as potentially more dangerous than the Santa Monica-Hollywood fault, which has an estimated potential of generating a magnitude 6 (Richter) earthquake with a recurrence period of 10,000 years. Although the effects of a major seismic event on the reinforced concrete buildings surrounding the reactor are uncertain, it will be assumed for this discussion that such an event is capable of collapsing one or more of those buildings. Furthermore, both the Los Alamos study and the Safety Evaluation Report assumed that a collapse of the reactor building could result in a collapse of the reactor biological shield and the reactor core. Two effects of the hypothesized reactor collapse resulting in a crushed core have been investigated.

In the Los Alamos investigation it was assumed that up to the immediate moment of the reactor's collapse the reactor had been operating continuously at full power (100 kw) for a sufficiently long period of time (months to years) to reach a near-equilibrium fission product inventory. Los Alamos examined whether reduced convective heat transfer in a collapsed configuration of the core could lead to core melting by decay heat accumulation. The Los Alamos study concluded that in such circumstances the core could not melt and fission product release by that mode was not possible. It should be noted that the reactor has never been operated under conditions that would result in attaining full power fission product equilibrium. The UCLA reactor operates at an annual average power level of less than 5 kw; the long term historical average is approximately 3 kw.

Based on the Battelle study which considered a core crushing event, the SER assumes that seismically-induced core damage could sever fuel plates and release fission products to the environment. In the SER it is calculated that atmospheric dispersion of the radioiodines in the crushed core situation could yield an estimated thyroid dose of 30 rem to individuals at the boundary of the demolished reactor room.

When considering the credibility of any core crushing scenario, it should be recognized that the reactor is a dense concrete and graphite structure. The thick, short spans of reinforced concrete blocks above the reactor have enormous compressive strength relative to any conventional building structure. It is by no means certain that the reactor core would be crushed in the event of the collapse of the reactor building.

During periods of major core maintenance, the core may be exposed and more vulnerable to a major seismic event. Core maintenance at the UCLA facility occurs no more frequently than once in five years. In order to minimize radiation exposure of personnel, core maintenance is not begun until three weeks after the last shutdown. At that time the core is exposed and the fuel unloaded in a single day, any required maintenance is performed, and subsequently the fuel is reloaded and the core covered in a single day. The fuel is not in the reactor while maintenance is in progress. The period for which the reactor is both exposed and at least partially loaded is no more than 16 hours during any five-year period. Without speculating on the probability of an open, partially loaded core and the simultaneous occurrence of a seismic event severe enough to collapse the reactor building and crush the core, it can be pointed out that the radiological releases postulated for such a case would not be quantitatively different from those postulated for the closed core cases.

8.3.2 CATASTROPHIC SEISMIC EVENT WITH FLOODING

The Battelle study considered the possibility of a criticality-type accident in the event there occurred a catastrophic rearrangement of the core with subsequent flooding of the reactor. Battelle assumed that a shock sufficient to produce the precise structural rearrangements of the core needed for a criticality accident would also lead to a loss of the existing reactor water due to the severing of the water lines. In such a case the water would have to be replenished from some source to restore the moderator necessary for such an accident.

It is conceivable that subsequent flooding of the reactor room could occur as the result of earthquake-induced failure of the Stone Canyon Reservoir which is positioned in the hills to the north of the UCLA campus [14,15]. If the dam were to fail, a portion of the UCLA campus would be flooded. The flood resulting from instantaneous dam failure is hypothesized to be of a magnitude capable of destroying a substantial part of west Los Angeles [14].

In the absence of core crushing, flooding alone will not produce fission product releases. Various scenarios were considered in the Battelle study which assumed a critical reactor, structural rearrangement of the core or stuck control blades, and loss of water with subsequent replenishment of the water-moderator by flooding. Battelle found that structural rearrangements of the core into some more optimal geometry of reduced minimum critical mass and large excess reactivity was not credible and, it may be added here, appears to imply some interpenetration of graphite and fuel while ignoring the intervention of the cadmium control blades. Moreover, the reactor is considered to be near optimally moderated in the sense that additional moderation drives the reactor less critical. It has long been known that wetting of the graphite results in a loss of excess reactivity, an effect which could alternatively be described as loss of reflector efficiency. Flooding beyond the optimum moderation level would be expected to lower the system reactivity.

Accepting the assumptions of the SER for the case of core crushing, subsequent flooding of the reactor could result in the dispersion of fission product releases in the flood water, which would be expected to disgorge to the Pacific Ocean southwest of the UCLA campus.

8.3.3 GRAPHITE FIRE

The Battelle study considered a general building fire as an initiator of an accident, but discounted the credibility of the cause except for the case of a fire fueled by reactor materials or other combustibles which might be used in the reactor room. The initiation of a reactor accident by a fire external to the reactor room is not credible in that the reactor building and the surrounding complex are constructed of reinforced concrete.

The principal combustible material routinely present in the reactor room is graphite. Ordinarily the graphite is contained within the reactor. During major core maintenance, which occurs only rarely, the reactor graphite may be stacked outside of the reactor. There is also a graphite sigma pile of approximately 64 cubic feet in the northeast corner of the reactor room. Small amounts of other combustibles, such as wood, cloth and paper products, are often present in the reactor room. During major core maintenance organic solvents may be brought in for decontamination purposes. Borated paraffin blocks, which are not readily combustible, are used as additional shielding.

Battelle noted that the plausibility of a graphite fire within a reactor core or thermal column enclosed in concrete shielding is limited by the available oxygen supply. However, Battelle assumed an air flow rate through an Argonaut reactor of 250 cubic feet per minute. The air flow rate through the UCLA reactor is actually less than 100 cubic feet per hour, approximately 0.7% of the rate assumed by Battelle. Under the UCLA conditions, it is much more likely that any graphite fire that managed to get started would suffocate due to lack of air and the buildup of combustion products. Battelle also discussed the possibility of a graphite fire occurring when the core is exposed. However, when maintanance is performed on the graphite or other elements of the core, the fuel is not in the core. Any scenario involving an open core, fully or partially fueled, unattended, with the graphite exposed and in contact with a substance capable of causing graphite ignition is not credible.

8.3.4 REACTIVITY INSERTION ACCIDENTS

It may be assumed that the investigations of Battelle and Brookhaven were designed to set a conservatively safe limit upon the excess reactivity to be permitted in an Argonaut reactor. The Battelle investigation concluded that melting and consequent fission product release would not occur with the rapid addition of excess reactivity in the amount of 2.6%. Their choice of a prompt neutron life time of 1.4×10^{-4} seconds adds an element of conservatism to the calculation because this parameter is at least 1.9×10^{-4} seconds for Argonaut reactors.
Brookhaven examined the ramp insertion of $3.00 of excess reactivity defined with $\beta = 0.00714$ (effective delayed neutron fraction). UCLA uses $\beta = 0.0065$ and the same excess reactivity would be termed $3.30. By either definition, the excess reactivity is approximately 2.14%. The Brookhaven effort was aimed at examining the safety of this amount of reactivity. The study confirmed that the ramp insertion of 2.14% excess reactivity is safe, but no conclusions were drawn concerning the maximum safe upper limit.

Neither Battelle nor Brookhaven addressed the question of how such a large reactivity insertion could occur. Brookhaven did suggest that one or more large cadmium sleeves having a total negative reactivity on the order of $3.00 inserted in a vertical port might fall out of the reactor. A negative reactivity of this worth is conceivable. But the Brookhaven study does not suggest why such an object would be introduced into the reactor, nor how once introduced, it could be made to deviate from the normal gravitational forces, and fall "up" and out of the reactor. As postulated, the event is not credible.

8.4 FUEL-HANDLING ACCIDENT

UCLA has adopted the fuel-handling accident proposed in the Battelle study as the most credible accident and has used this event for the purpose of planning emergency responses. However, in extending that generic study to the UCLA circumstances, three modifications are in order. First, although Battelle assumed 365 days of continuous operation at 100 kw, the UCLA reactor operates an average of less than 5% of that time. Continuous operation is typically no more than four hours at a time and only very rarely more than eight hours. To the knowledge of the current staff, the reactor has never operated continuously for more than 24 hours; the last occasion of 24 hour continuous operation occurred in 1974. Second, Battelle assumed that the fuel handling accident occurred immediately upon shutdown at the end of the 365 days. However, a holding period of three weeks is observed at UCLA to reduce the potential of radiation exposures to the staff. Third, although the NRC review assumed that the fuel area exposed in the core collapse event would be equivalent to one entire fuel bundle, in a fuel-handling accident the denuding of a single fuel plate is considered to be the worst possible credible consequence.

8.4.1 CORE INVENTORY

The twenty years of UCLA reactor operations have generated a cumulative energy of 19.4 megawatt days, approximately half of the total energy assumed to be generated in one year in the generic study. Most of the historically generated energy was produced in the last seventeen years and the long term average energy generation is approximately 27.4 megawatt hours per year or an average power level of about 3 kw. The most intensive years of operation were in the middle 1960's, and the most intensive three-month interval identified in that era was the fourth calendar quarter of 1966. The energy generated in that quarter was 17.5 megawatt hours for an average power level of 8 kw.

The statistical history is relevant because long lived isotopes such as Krypton-85 accumulate slowly over a long period of time. Isotopes of intermediate life (Iodine-131 and Xenon-133) are present in quantities reflecting the prior several months of operation. The inventory of shorter lived isotopes depends upon the most recent operational history of the reactor. However, these generalizations must include consideration of precursor decay, particularly for any short lived gaseous isotope that arises as a decay product of a longer lived reactor [10]. Subsequent to the release of the SER, Brookhaven National Laboratory released a related analysis of Argonaut reactors [11]. Although differing substantially in scope and focus, these studies and reviews reached similar conclusions.

The Battelle study examined a broad spectrum of accident potentials including large reactivity insertions, core crushing, flooding, fire and fuel handling. The Battelle investigators concluded that the only credible accident that would result in significant radiological releases was a fuel-handling accident.

The Los Alamos study examined the properties of a crushed core and found that the altered configuration would not subject the core to melting by radioactive decay heat under the reduced convective cooling conditions within the crushed core. The Los Alamos investigator reported that even after long-term continuous operation of the Argonaut reactor at 100 kw, the maximum fuel temperature (following shutdown) in a core-crushing episode was calculated to be 358°C, well below the aluminum-uranium alloy melting point.

A transient analysis of the Argonaut reactor was conducted by the Brookhaven National Laboratory using computer modeling. The Brookhaven report concluded that a rapid ramp insertion of excess reactivity would not drive the peak core temperatures to the melting point, a conclusion in qualitative agreement with the Battelle finding.

All of the studies were concerned with accidents which might lead to radiological consequences. Specifically examined are the possibility that release of radioactive material due to core melting could be brought about by excess reactivity insertions (Battelle or Brookhaven) or by core crushing (Los Alamos). Each of the investigations determined that core melting was a non-credible event and that fission product release by this mode is not a credible consequence of an Argonaut reactor accident. Battelle, Los Alamos, and the SER found that some fission product release could result in the case of a mechanically damaged and crushed core. Battelle noted that flooding of the core during or shortly after crushing would result in some release of fission products to the flood waters. It was generally assumed that the core crushing scenario could be produced by a major seismic event, although neither the probability of such an event nor the proposed mechanism of the crushing was examined by any of the investigators. Battelle discussed the possibility of a graphite fire and found that it would not create sufficient damage to melt any fuel or initiate a metal-water reaction [8 - Abstract].

Among the various accident potentials considered by these investigators, a fuel-handling accident was found to be the most credible accident that might result from ordinary facility operations, as distinguished from accidents which might be initiated by catastrophic natural events. Accordingly, a fuel-handling accident has been adopted as the design basis accident for reactor emergency planning.

precursor.

An extended holding time prior to core entry for fuel transfer is conventionally practiced because of the relatively modest shielding (6 inches of water) that remains after removal of the sixty inch concrete biological shield and a twelve inch lead and graphite plug. A holding time of three weeks was observed in the 1974 core entry and led to acceptable personnel radiation dosages in the subsequent core entry and fuel-handling operations. The holding time can also be regarded as an accident control parameter and it is appropriate to demand a minimum holding (non-operating) period of three weeks prior to any fuel-handling operation (Technical Specification 3.6.3.4).

Therefore, the following operational schedule will be assumed.

a. Operation for two or more months at an average power level of 15 kw. (That average level is five times the historical long term level and approximately twice the highest intensity identified in any quarterly period.)

b. A final run of 24 hours at 100 kw.

c. A holding period of three weeks prior to core entry for fuel-handling.

Note that this operational schedule is considerably more intensive than is suggested by operating experience and any short period excesses can be limited by a restriction of operational intensity to less than 2.0 megawatt hours in any consecutive seven day interval (Technical Specification 3.8.3.C). This condition supplements and does not replace the existing limiting of 5% of the total potential of 8760 full power hours in any consecutive 365 day period.

8.4.2 LOSS OF CLADDING AND FISSION PRODUCT INVENTORY OF INTEREST

The 264 rolled fuel plates used in the UCLA reactor consist of an aluminum cladding tightly bonded to the aluminum-uranium core "meat". No examples of "peeling" could be identified by the University of Michigan where similar fuel had been handled frequently at much higher burnup than UCLA can ever expect to realize [21].

Aluminum is readily attacked by acids and alkalies. Such chemicals are not used in connection with reactor operations and are not stored or used for any other purpose in the reactor room. The presence in the reactor room of a tub or vat containing such chemicals, and of sufficient size to immerse a fuel element is not credible.

Aluminum metal is highly malleable and ductile, and hence deforms rather than shatters under impact. A loss of cladding accident would require abrasion, shearing, or tearing forces, and no specific event has been proposed to describe the mechanism of such damage. The greatest conceivable area which could be exposed would result from the complete removal of the cladding from the two exterior flat surfaces of a fuel element. The postulated exposed area is equivalent to that of a single fuel plate, and therefore the inventory of interest is that of a single plate. If we assume that the inventory of the most active element or plate is 50% greater than that of the average plate, then the fraction of the total inventory which will be present in the most active plate in the core is:

$$f = \frac{1 \times 1.5}{264} = 5.7 \times 10^{-3} \text{ or } 0.57\%.$$

Using this fraction, the inventory of gaseous fission products of interest is shown in Table III/8-1.

Except for Krypton-85, the entries are those identified in [8], which remain in significant quantity after the 21 day holding period. The Krypton-85 is approximately 1/11 of the Battelle value and is close to the equilibrium value that would be approached in forty years of operation at an average power of 5 kw.

8.4.3 FISSION PRODUCT RELEASE

In discussing the possible release of fission products in a fuel handling accident, Battelle assumed that cladding removal would release all gaseous fission products within one recoil length of the exposed surface. The fission fragment recoil length in aluminum is approximately [19]

$$1.36 \times 10^{-3} \text{ cm}.$$

The prompt release of fission products from unclad fuel elements has been discussed theoretically by Olander [16]. Within the operating reactor with fission events in progress, the prompt release of a fission fragment can occur only if the fragment is formed within one recoil length of the surface of the fuel element. For specific fragments created at uniformly distributed sites within one recoil length of the surface, Olander shows that only 1/4 of those fragments will be emitted in directions which will carry them to this surface. The other fragments remain trapped in the fuel matrix. The prompt release terminates when the reactor is shut down.

For a fuel plate with a cladding thickness greater than one recoil length, the cladding can be expected to absorb and trap almost all of the fission fragments emitted from the fuel meat. The subsequent release of embedded fission products will be governed by diffusion rates in the solid matrix of fuel meat or cladding. Fission fragment diffusion in aluminum and aluminum-uranium alloys has been the subject of a number of investigations [17,18,20]. The rates are extremely slow at room temperature, and significant releases are observed only if the material is raised to a temperature of 400°C or higher.

Table III/8-1 Inventory of One Fuel Plate Containing 0.57% of the Core Inventory, Curies

Nuclides	At Shutdown	At 21 Days
Kr-85	0.09	0.09
Xe-133	5.98	0.62
I-131	2.83	0.49
I-132	5.81	0.07
I-133	20.20	-

Inasmuch as the hypothetical fuel handling accident occurs long after active fissioning has ceased, an assumed release of all of those gaseous fission products formed within one recoil length of the surface, yields a highly conservative overestimate of the expected release. The release fraction is two recoil lengths (2.74×10^{-3}cm) divided by the fuel matrix thickness (0.102 cm) or 2.7%.

The reactor room is not a sealed structure, hence the common practice of attempting confinement by shutting down the ventilation system is inappropriate. Table III/8-2 shows the release to the reactor room, the concentrations in the room and at the stack exit, and the personnel dose in the room under the assumption that the ventilation system continues to withdraw 9000 CFM from the reactor room and exhaust 14000 CFM at the stack exit. The entire release and sweep-out is assumed to occur in one hour. The consequences are not sensitive to the rate of release, but do depend upon the amount of material released.

8.4.4 CONCENTRATION AND DOSE STANDARDS

The thyroid uptake of radioiodines leads to a cumulative dose. The calculable dose for an exposure to concentration C for time T is proportional to the product C•T and the same dose results if the concentration is doubled and the exposure time is cut in half.

The maximum permissible concentration (MPC) of iodine-131 for the general public is 10^{-10} microcuries per milliliter [24]. Iodine-131 is the longest lived of the iodines considered here and the permissible concentration is for continuous exposure. 10 CFR 20-106a permits annual averaging and implicitly, an annual dose limit. An exposure of one hour per year to a concentration of 8760 times MPC will produce the same cumulative dose as continuous exposure to one MPC for one year. Thus, the permissible concentration for an exposure of one hour, occurring no more frequently than once per year, is $8760 \times 10^{-10} = 0.876 \times 10^{-6}$ microcuries per milliliter. The iodine-131 concentration in the stack effluent, resulting from a fuel handling accident (Table III/8-2) is approximately 64% of the permissible one-hour, once per year release. Consequently, the exhaust stack plume cannot expose anyone to a thyroid dose greater than that which would result from continuous exposure for one year to the permissible concentration of 10 CFR 20, Appendix B, Table II, column 1.

Tighter standards have been developed to define Emergency Action Levels (EAL's) for the purpose of emergency preparedness [22]. At the lowest EAL, the standard requires notification of the Commission if a release exceeds 10 times MPC when the concentration is averaged over 24 hours. Equivalently, the Commission is to be notified if a release of one hour duration has a concentration exceeding 240 times MPC. Whether treated as 24 hours at 10 times MPC or one hour at 240 times MPC, the thyroid dose due to iodine-131 under this standard is the same and is less than 3% of the dose that would accumulate in one year of continuous exposure to the maximum permissible concentration of iodine-131.

Table III/8-2 Releases, Concentrations and Dose In Reactor Room for Release of 2.7% of the Gaseous Fission Products in One Fuel Plate

	Release Curies	Concentration, Ci/m^3		Dose In Room, REM
		In Room	At Stack	
Kr-85	0.0024	0.16×10^{-6}	0.10×10^{-6}	--
Xe-133	0.0170	1.10×10^{-6}	0.71×10^{-6}	0.2×10^{-3}
Total Whole Body Dose Equivalent From Nobel Gases				0.2×10^{-3}
I-131	0.0130	0.87×10^{-6}	0.56×10^{-6}	1.57 (thyroid)
I-132	0.0020	0.13×10^{-6}	0.08×10^{-6}	0.01 (thyroid)
Total Thyroid Dose Equivalent From Radioiodines				1.58

8.4.5 SITE BOUNDARY AND AREA OF RADIOLOGICAL CONCERN

The emergency preparedness standard referred to above applies to concentrations at a site boundary and attempts to distinguish between on-site personnel and the general public beyond the boundary. The site boundary also encloses the area controlled by a licensee, and therefore distinguishes between on-site and off-site forces and resources available to a licensee for dealing with emergencies. For this last purpose, the site boundary is the campus boundary. However, the UCLA boundary is not fenced and no clear distinction can be made between the campus population (faculty, students, staff, and visitors) and the general public. More significantly, the boundary is far from the reactor and concentrations at the boundary due to radiological releases from the worst possible accident at the reactor will be extremely small.

The foregoing can be illustrated by noting that the campus boundary nearest the reactor is about 830 meters east of the reactor exhaust stack. Employing the Gaussian plume model with a stack radius of 0.41 meters, an air exit velocity of 12.5 meters per second and a wind speed of 3.5 meters per second [23], the plume center line concentration 830 meters downwind will be reduced by a factor of 2000 from the value at the stack. When averaged over 24 hours, the concentration will be 13% of MPC and 1/75 of the lowest Emergency Action Level of 10 times MPC.

The emergency preparedness standard may be used in reverse to define an area of radiological concern, beyond which the concentration will not exceed 10 times MPC when the release is averaged over a 24 hour period. Using the previous parameters and the stack concentration from Table III/8-2, the calculated boundary is a circle, one meter in radius surrounding the stack. The ninth floor (roof) of the Mathematical Sciences Addition is an unrestricted area and the closest approach that the general public may easily make is about 8.2 meters from the stack. The dilution factor at that distance is approximately 160, and therefore the 24 hour averaged concentration at that point is about 1.3 times MPC or 1/6 of the criterion for invoking the lowest Emergency Action Level.

8.3.6 CONCLUSIONS

The radiological consequences of the worst credible accident will not expose any member of the general public to concentrations in excess of permissible limits. Further, the projected exposures are not at the level sufficient to require Notice of an Unusual Event under the applicable emergency preparedness standard [22].

10. REFERENCES

1. R. D. MacLain, "UCLA Training Reactor Hazards Analysis, Final Report," Department of Engineering, University of California, Los Angeles, Report No. 60-18, March 1960.

2. AMF Atomics and General Nuclear Engineering Corporation, "Start-Up Report AMF-GNE Educator Reactor, University of California, Los Angeles, January 1961.

3. American Society of Mechanical Engineers, Nuclear Reactor Plant Data, Research and Test Reactors, Volume 2, March 1959, McGraw-Hill Book Company, New York, New York.

4. C. E. Ashbaugh, "Nuclear Energy Laboratory - Report," School of Engineering and Applied Science, University of California, Los Angeles, 1977.

5. A. L. Babb, "University of Washington Training Reactor Final Hazards Summary Report, College of Engineering, University of Washington, Seattle, Washington, May 1960.

6. Personal communication with Pat Miller, Reactor Supervisor, University of Washington, January 1980.

7. Personal communication with N. J. Diaz, Reactor Supervisor, University of Florida, January 1980.

8. S. C. Hawley, et al., "Analysis of Credible Accidents for Argonaut Reactors," NUREG/CR-2079, April 1981.

9. G. E. Cort, "Fuel Temperatures in an Argonaut Reactor Core Following a Hypothetical Design Basis Accident (DBA)," NUREG/CR-21[illegible], June 1981.

10. "Safety Evaluation Report Related to Renewal of the Operating License for the Research Reactor at the University of California at Los Angeles, July 1981. Available through the U. S. Nuclear Regulatory Commission undet Docket No. 50-142, License R-71.

11. Brookhaven National Laboratory memorandum Neogy to Carew, "Transient Analysis of Argonaut Reactors," November 28, 1981.

12. R. Greenfelder, "Maximum Credible Rock Acceleration From Earthquakes in California. Map Sheet 23," California Division of Mines and Geology, 1972. Revised, August 1974.

13. Lindvall, Ricter & Associates, "Seismicity, Geology and Design: Earthquake Report for Stone Canyon Dam," March 1976.

14. Ayyaswamy, P. et al., "Estimates of Risks Associated with Dam Failure," UCLA-ENG-7423, March 1974.

15. City of Los Angeles, Department of Water and Power, "Stone Canyon Dam Stability Evaluation," Report AX-213-34, March 1977.

16. D. R. Olander, "Fundamental Aspects of Nuclear Reactor Fuel Elements," Technical Information Center, TID-26711-PI, 1976, p. 289-293.

17. D. S. Billington, "Radiation Damage in Reactor Materials," Peaceful Uses of Atomic Energy, Proceedings of the International Conference in Geneva, August 1955, Volume 7, p.425.

18. M. T. Simnad and L. R. Zumwalt, "Diffusion of Fission Products (and Transmutation Products) in Non-Fuel Materials," General Atomic Division of General Dynamics, GA-6542, December 1965, p.76-77.

19. C. E. Weber and H. H. Hirsch, "Dispersion Type Fuel Elements," Peaceful Uses of Atomic Energy, Proceedings of the International Conference in Geneva, August 1955, Volume 9, p. 197.

20. J. A. L. Robertson, Irradiation Effects in Nuclear Reactors, Gordon and Breach, New York, 1969, p.262-264.

21. Reed R. Burn, Personal Communication. University of Michigan, January 19, 1982. Confirmed no degradation at the high burnup values shown in "Research, Training, Test and Production Reactor Directory," American Nuclear Society, 1980, p.1627.

22. "Standard for Emergency Planning for Research Reactors," ANS 15.16, Draft 2, November 29, 1981. (Available from the American Nuclear Society, 555 North Kensington Avenue, LaGrange Park, Illinois, 60525).

23. Letter, Wegst (UCLA) to Tedesco (USNRC), September 5, 1980.

24. Title 10, Chapter 1, Code of Federal Regulations, Part 20, Appendix B, Table II, Column 1.

APPENDIX III

ARGONAUT SAFETY ANALYSIS REPORT (ASAR)

Attachment A

Analysis of Credible Accidnets

for Argonaut Reactors

NUREG/CR-2079-PNL-3691
Battelle Pacific Northwest Laboratory
Richland, Washington
April 1981

[incorporated by this reference]

APPENDIX III

ARGONAUT SAFETY ANALYSIS REPORT (ASAR)

Attachment B

Fuel Temperatures in an Argonaut Reactor Core

Following a Hypothetical Design Basis Accident (DBA)

NUREG/CR-2198, by G. E. Cort
Los Alamos National Laboratory
Los Alamos, New Mexico
June 1981

[incorporated by this reference]

APPENDIX IV

Emergency Re.,ponse Plan

This section is reserved.