UNIVERSITY OF FLORIDA TRAINING REACTOR ANNUAL PROGRESS REPORT

SEPTEMBER 1, 2004 - AUGUST 31, 2005

Submitted by
Dr. William G. Vernetson
Director of Nuclear Facilities

Department of Nuclear and Radiological Engineering
College of Engineering
University of Florida
Gainesville, Florida

February 2006

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2004–5 ANNUAL REPORT

I. INTRODUCTION

The University of Florida Training Reactor's overall utilization for the past reporting year (September 2004 through August 2005) continued to be at historically high levels of quality usage, limited only by unavailability of the reactor or necessary personnel. It was a productive year considering that there were no excessively large outages that hampered reactor usage throughout the year until the trip on a failed temperature monitoring channel in late June 2005, with the outage extended to most of the remainder of the reporting year and into the next year partly due to other equipment failures. The diversity of users and usages continues to rank among the best in the history of the facility, especially considering that availability this past reporting year was nearer to historical levels at nearly 74% after 85% last year but being down to 36.5% in the previous 2002–3 year after less than 35% and 59% the previous two years after the 1999–2000 reporting year's value at over 88%. The relatively good availability for the 2004–5 reporting year was primarily due to having no excessive outages until late in the year with the longest forced outage of the reporting year being 18⅛ days in June/July/August 2005 for the failure in the temperature monitoring system extending to the next reporting year. Other significant outages were for designing and installing a new device for checking the primary coolant level trip in March 2005 (4 days), to clean and repair the device for secondary flow trip on well water flow in July 2005 (8 days), to replace a failed period trip bistable card in July/August 2005 (16⅞ days) and to repair a leak in the secondary heat exchanger sample line in August 2005 (3⅜ days) with the latter three failures effectively extending the outage for the temperature monitor system failure as the primary cause for the extended outage at year's end. There were also two administrative outages in September 2004 for hurricane watches (7½ days) plus a lengthy planned outage (6½ days) from normal operations for the annual calibration of nuclear instruments in March 2005. Unlike in years prior to 1990–91, this availability accounts for lost availability for administrative reasons as well as for repair and maintenance related reasons.

The University of Florida Training Reactor (UFTR) continues to experience a high rate of utilization in a broad spectrum of areas with total utilization continuing near the highest levels recorded in the early 1970s and most usage indicators remaining high with quality usage occurring whenever system and operator availability permits. This broad-based utilization has been supported by a variety of usages including research and educational utilization by users within the University of Florida as well as by other researchers and educators around the State of Florida through the support of the Department of Energy (DOE) Reactor Sharing Program and several externally supported usages. A number of science fair projects were also accommodated. Less effort than usual has also been devoted to facility enhancement except when necessary; a key ingredient accounting for this situation has been the lack of a full-time Reactor Manager/SRO in place for the entire year. During this 2004–5 year one part-time SRO who graduated in April 2004 continued to be employed until he obtained permanent employment in the utility industry upon leaving on March 22, 2005. One part-time student operator trainee continued employment throughout the year and was finally licensed/certified as a Reactor Operator on August 22, 2005 as one other student ex-Navy operator worked intermittently until leaving employment in late December 2004 of this reporting year. Personnel associated with the UFTR are listed in Chapter II; this does not include NAA Laboratory personnel except where also involved with UFTR operations. The loss of the most experienced NAA laboratory assistant at the beginning of the 2002–3 reporting year continued to present a challenge throughout the reporting year for research usage of the facility. One assistant was lost in midyear with another hired for the entire reporting year as the primary non-licensed support technician. Several part-time assistants were also hired in the latter half of the year (March and June 2005) but were becoming productive at year's end as their training continues.

The package to apply for UFTR relicensing was submitted with a cover letter dated July 29, 2002 to allow the UFTR R-56 license to remain effective until action is taken on the relicensing submittal. The NRC letter acknowledging the UFTR license renewal and continued effectiveness of the R-56 license as a "timely" renewal application is dated August 26, 2002. Some errors were noted primarily due to computer formatting and retrieval errors made during the document conversion process for duplication (printing) of the Final Safety Analysis Report (FSAR). There were no actual changes to the FSAR content so these changed pages were provided to the NRC with a cover letter dated February 23, 2003. Though NRC had indicated they had begun to review the submission, there had been no other official response until the request for additional information (RAI) was received on April 11, 2005 with a cover letter dated April 5, 2005 of this reporting year. Because of the DOE decision to move forward on HEU-to-LEU conversion the NRC indicated it would discuss the timing of the responses to the questions. Questions not affected by the conversion may have answers delayed until after conversion is complete; those affected are to be answered with the HEU-to-LEU conversion submittal.

The remaining chapters of this report have contents as described below. As noted above, Chapter II summarizes University of Florida personnel associated with the reactor including those employed by the facility itself, primary support personnel from the Radiation Control Office, membership of the Reactor Safety Review Subcommittee as well as personnel in line responsibilities for UFTR administration and for the Radiation Control Office. Unlike in the 2003–4 reporting year, the Level 1 administration of the UFTR facility was not changed; however, the Chair of the RSRS had been replaced with the alternate in November 2003 after the Chair became ill and retired from the University. On December 16, 2004 a permanent change was made as Dr. David E. Hintenlang became Chair of the RSRS as the only significant administrative change during the 2004–5 reporting year.

Chapter III summarizes key aspects of UFTR facility operation including Reactor Sharing Program users. Table III-1A is a list of such user institutions and Table III-1B provides some details on the usage. Energy generation is listed in Table III-2, key-on time, run time and availability in Table III-3, availability and causes of unavailability in Table III-4 as well as unscheduled (three) and scheduled (one) trips in Tables III-5A and III-5B. The log of unusual occurrences constitutes Table III-6 and contains eight items for 2004–5. Though no events are considered to have compromised reactor safety or the health and safety of the public or facility personnel, the eight occurrences described in Table III-6 are the most significant events for the 2004–5 reporting year. Included in Table III-6 are the three trips noted in Table III-5A.

Chapter IV contains a listing and description of all modifications and/or changes in conditions made to reactor-related facilities during the reporting year. Only two items are included with a 10 CFR 50.59 package prepared for all entries (some carried over from the previous reporting year) with none evaluated and determined to require NRC approval prior to implementation.

Chapter V contains a general introductory description of maintenance, tests and surveillances of UFTR reactor systems and facilities undertaken during the reporting year. Table V-1 is a chronological tabulation and description of all scheduled UFTR surveillances, checks and tests performed on a quarterly or less frequent basis. Table V-2 then contains a chronological tabulation of UFTR preventive and corrective maintenance actions performed during the reporting year.

Chapter VI contains descriptions of changes to Technical Specifications, FSAR, Emergency Plan, Standard Operating Procedures and other significant documents. During the 2004–5 reporting year there was one change to the Tech Specs, the first since Technical Specification Amendment 23 was approved and implemented in the 2001–2 reporting year. Technical Specification Amendment 24 affecting pages 19 and 21 was submitted to NRC with a letter dated September 17, 2004. This amendment extended the fuel inspection interval and the interval for mechanical integrity inspection of the control blades and drive system to ten (10) years at intervals not to exceed twelve (12) years from the previous specification of five (5) years not to exceed six (6) years. Tech Spec Amendment 24 was implemented in January 2005. The relicensing package included various updated documents including the Technical Specifications, FSAR, Emergency Plan and Requalification and Recertification Training Program. This document submittal was accepted for review by the NRC in August 2002 with no action expected for several years. There were also no changes to the FSAR though the proposed FSAR submitted for relicensing was discovered to have some errors primarily due to computer formatting and retrieval errors made during the document conversion process for duplication (printing) of the FSAR. There were no actual changes to the FSAR submitted for relicensing so these changed pages were provided to the NRC with a cover letter dated February 23, 2003 of the previous reporting year. This package is available for review at the UFTR facility. Though NRC had indicated they had begun to review the submission, there had been no other official response until the request for additional information (RAI) was received on April 11, 2005 with a cover letter dated April 5, 2005. Because of the DOE decision to move forward on HEU-to-LEU conversion, the NRC indicated it would discuss the timing of responses to the questions. Most questions not affected by the conversion may have answers delayed until after conversion is complete.

Revision 12 to the UFTR Emergency Plan was submitted in August 2001 and fully implemented in February 2002 of the 2001–2 reporting year with no changes made during the 2004–5 or the previous two reporting years. A revised ALARA program was generated during the 2002–3 reporting year with no changes this year. There were also no changes to the UFTR Physical Security Plan or to the Respiratory Protection Program during the 2004–5 reporting year. The UFTR Biennial Reactor Operator Requalification and Recertification Training Program was submitted for renewal with minor changes in May 2005 for the July 1, 2005–June 30, 2007 cycle. The only other significant reactor-related document changes in the 2004–5 reporting year involved changes to various Standard Operating Procedures. One new procedure was generated during the 2004–5 reporting year with no procedures being revised during this reporting year as a result of periodic reviews. In addition only three temporary change notices were implemented as this was a relatively inactive year in this area.

Finally, Chapter VII contains a review summary of radioactivity released and environmental surveillances performed. Releases described include gaseous Argon-41 and liquid waste released at activity levels below the lower limit of detection with no solid waste shipments. Chapter VII also contains a summary of environmental monitoring performed using Luxel dosimeters including a breakdown by month. Again, all environmental dose results are essentially negligible. The last section shows a summary of personal radiation exposure for facility personnel and several visitors with all exposures well below regulatory limits.

More details in each of these areas are contained in the following six chapters. If additional information is required, the facility may be contacted.

The expectations for the 2005–6 reporting year are very positive. Significant opportunities for expanded education and research usages are apparent. The possibilities for continued growth in existing and new program areas are a challenge that continues to be addressed following the return to historically expected availability during the 2003–4 reporting year especially noting resource limitations, pending license renewal, anticipated HEU-to-LEU fuel conversion, having no permanent Reactor Manager and the need to license additional operators as well as continue training part-time students to develop and maintain expertise in the NAA Laboratory. Nevertheless, with sufficient support, there is no limit to possibilities for growth in facility usage.

II. UNIVERSITY OF FLORIDA PERSONNEL ASSOCIATED WITH THE REACTOR

A. Personnel Employed by the UFTR

W. G. Vernetson	–	Associate Engineer/Director of Nuclear Facilities and Senior Reactor Operator (September 2004 – August 2005)
B. Shea[1]	–	Technician and Senior Reactor Operator (3/4 time) (September 1, 2004 – March 22, 2005)
M. Berglund	–	Student Technician and Senior Reactor Operator Trainee (1/3 time) (September 2004 – February 2005) (3/4 time) (March 2005 – August 21, 2005)
	–	Student Technician and Reactor Operator (3/4 time) (August 22, 2005 – August 31, 2005)
R. Leug	–	Student Technician and Senior Reactor Operator Trainee (1/20 time) (September 2004 – December 2004)
M. Holman	–	Graduate Project Student (1/2 time) (September 2004 – December 2004)
	–	Graduate Student Reactor and Laboratory Technician/HP Assistant (1/4 time) (February 2005 – March 2005) (3/4 time) (May 2005 – August 2005)
D. Seifert		Secretary (September 2004 – August 2005)
M. Crawford, G. Joseph, N. Patel, C. McJunkin, M. Yenatskyy, B. Stewart, A. Chomat	–	Student Technicians for various parts of the year usually working in NAA Laboratory but effectively providing approximately 1/4 time commitment to reactor related activities

[1] B. Shea graduated with a bachelor's degree on April 30, 2004. He resigned his position effective May 28, 2004 but continued employment in this reporting year until he left for a permanent position in industry in late March 2005.

B. Radiation Control Office

D. L. Munroe[2]	–	Radiation Control Officer (September 2004 – August 2005)
J. J. Parker	–	Radiation Control Technician (September 2004 – August 2005)

Basic routine health physics is performed by UFTR staff; however, assistance from the Radiation Control Office is required for operations where a significant dose (Level I RWP) is expected or possible and where certain experiments are inserted or removed from the reactor ports. These personnel are also required for certain operations where high contamination levels may be expected such as fuel inspection activities or core area maintenance activities. They also periodically review routine UFTR radiation control records and operations and assist in performance of certain radiation safety and control related surveillances. Several others with only infrequent contact at the UFTR are not listed though they are available for backup purposes or if an emergency should arise.

C. Reactor Safety Review Subcommittee (RSRS)

W.S. Properzio	–	RSRS Chairman (Specified Alternate Chair) (Associate Professor, Department of Environmental Engineering Sciences/Director, Environmental Health & Safety Division) (September 2004 – December 16, 2004)
D. E. Hintenlang	–	Member (Associate Professor, Department of Nuclear and Radiological Engineering) (September 2004 – December 15, 2004)
	–	RSRS Chairman (Associate Professor, Department of Nuclear and Radiological Engineering) (December 16, 2004 – August 31, 2005)
W. G. Vernetson	–	Member (Director of Nuclear Facilities/Associate Engineer, Department of Nuclear and Radiological Engineering) (September 2004 – August 2005)
D. L. Munroe	–	Member (Radiation Control Officer/Assistant Director, Environmental Health & Safety Division) (September 2004 – August 2005)
J. S. Tulenko	–	Member (Professor, Department of Nuclear and Radiological Engineering) (September 2004 – August 2005)

[2] The specified alternate for the RCO position is G.I. Snyder.

A. Haghighat	–	Member (Professor/Chairman, Department of Nuclear and Radiological Engineering) (September 2004 – August 2005)
D. P. Butt	–	Member (Associate Professor, Department of Materials Science and Engineering) (December 6, 2004 – August 31, 2005)
G. E. Sjoden	–	Member (Associate Professor, Department of Nuclear and Radiological Engineering) (December 6, 2004 – August 31, 2005)

D. Line Responsibility for UFTR Administration

J. B. Machen		President, University of Florida (September 2004 – August 2005)
P. P. Khargonekar	–	Dean, College of Engineering (September 2004 – August 2005)
A. Haghighat	–	Chairman, Department of Nuclear and Radiological Engineering (September 2004 – August 2005)
W. G. Vernetson	–	Director of Nuclear Facilities/Acting Reactor Manager (September 2004 – August 2005)

E. Line Responsibility for the Radiation Control Office

J. B. Machen	–	President, University of Florida (September 2004 – August 2005)
J. E. Poppell	–	Vice President, Administrative Affairs (September 2004 – August 2005)
W. S. Properzio	–	Director, Environmental Health & Safety Division (September 2004 – August 2005)
D. L. Munroe	–	Radiation Control Officer/Assistant Director, Environmental Health & Safety Division (September 2004 – August 2005)

III. FACILITY OPERATION

The UFTR continues to experience a high rate of utilization as total utilization continues at or near the highest levels recorded in the early 1970's in most areas when the reactor is available; with much higher availability during the first ten months of the 2004–5 reporting year, many indicators are up, some down for the year but with good results considering reduced availability of licensed operations staff. One SRO resigned in March 2005 and one of two operator trainees did so also; efforts were then made to license one individual who finally certified in August 2005. This continuation of a good rate of UFTR facility usage has been supported by a variety of usages ranging from research and educational utilization by users within the University of Florida to research, educational and training utilization by users around the State of Florida through the support of the U.S. Department of Energy's University Reactor Sharing Program with much of the costs of this latter usage not covered by Reactor Sharing. Again this year, several externally supported usages have also continued to impact reactor utilization and support the continued diversification of facility activities and capabilities, especially through the hiring of part-time laboratory assistants for support work in the analytical laboratory and to provide funding for facility improvements. For the eighth year in a row, there was also a Department of Energy University Reactor Instrumentation (URI) Program grant to provide support for instrumentation upgrades during the year as notice of such was received in June 2005.

As noted over the last nineteen years, the continuing refurbishment and upgrade of the Neutron Activation Analysis (NAA) Laboratory has impacted favorably on all areas of utilization from research projects using NAA to training and educational uses for students at all levels, especially for student design-related projects. With successful implementation of an improved remote sample-handling "rabbit" facility, efforts to advertise availability and encourage usage of the UFTR (especially for research) have proceeded in a favorable light, though always less quickly than hoped over the last seventeen years. Implementation of the standard rabbit capsule size with larger carrying capacity, the subsequent additional implementation of two state-of-the-art PC-based spectrum analyzer systems with complete ORTEC software packages for spectrum analysis and data reduction, the installation of an independent sample and standards drying facility as well as improved shielding around the pneumatic sample insertion (rabbit) system are all improvements that have been key factors in supporting facility usage by assuring an easier and faster turnaround of samples submitted to be irradiated for Neutron Activation Analysis. Current efforts continue to emphasize converting the NAA Laboratory to utilize computer-based analyzer systems based on Canberra software packages as they are considered more user-friendly with better support.

The Reactor Sharing usage of the reactor and NAA Laboratory facility continue to be a significant fraction of all usage. Table III-1A contains a listing of schools availing themselves of this opportunity, while Table III-1B contains brief summaries of the various usages. Some usages include trace element analysis of river sediments and other samples for researchers at Savannah State University as well as transmutation doping of isotopic Zn-68 foils for laser development research at the University of Central Florida. A number of science fair projects were also supported with good results at the state finals for students from Robert F. Munroe, Palmer Trinity, Lecanto and other high schools. Literally dozens of other class and small group educational and research usages were conducted for the various educational entities running the full range from the precollegiate level, such as PEEK Middle School Boys Camp, the GATORTRAX Workshop, Discovery Intermediate School Honors Students and Gainesville Country Day School, to Santa Fe Community College Radiography and Nuclear Medicine Technology students and teachers, Hillsborough Community College Nuclear Medicine Technology students, and many other similar groups including the Eye-on-Engineering High School Summer Camp which was particularly rewarding. A similar spectrum of on-campus users includes classes in Nuclear and Radiological Engineering, Environmental Science and Engineering, Reserve Officers Training Corps, Mechanical Engineering, and others.

Service usages included neutron transmission measurements on spent fuel pool absorber coupons for Holtec International as well as evaluation of trace elements in medicinal lanthanum carbonate for a Shands Hospital medical researcher.

Table III-2 contains a listing of energy generation by month for the reporting year. The yearly total of 19,901.03 kilowatt-hours energy generation is relatively low, partially due to not having sufficient licensed operators during much of the year and continued but not particularly because of having poor overall availability which was about 74% for the year, though very low over the final two months of the reporting year.

Table III-3 lists key-on time, experiment time, run time and availability for each month during the year. Values are very encouraging with nearly 314 hours of run time and a monthly average availability of only 74.10% despite poor personnel availability. Similarly, Table III-4 provides a detailed breakdown of availability/unavailability with primary causes of unavailability listed for each month of the reporting year. A fourth useful indicator is whether the unavailability is due to a forced outage, a planned outage, or for administrative reasons such as the two hurricane watches in September 2004 or the Christmas Holiday in December 2004. As noted, the relatively high availability this year was primarily due to relatively short forced outages until the last two months of the year, though administrative unavailability at 34⅞ days is higher than in any recent year.

Table III-5A lists and describes the three unscheduled trips for the year with minimal safety significance. Table III-5B lists one scheduled trip for the year which occurred during the administration of an NRC license examination at the request of the license examiner.

Table III-6 lists eight so-called unusual occurrences for the year with the three trips described in Table III-5A listed as three of these entries. Again, all eight have relatively low safety significance and essentially no impact on the health and safety of the public or reactor facility personnel.

TABLE III-1A

REACTOR SHARING PROGRAM
SUMMARY OF USAGE OF UFTR FACILITIES
(September 2004 – August 2005)

	School	Usages*	Faculty	Students
1.	Academy of Environmental Science Charter High School	2	1	5
2.	Alachua County Middle Schools *(Science Quest Workshops)*	2	3	48
3.	Alachua County Science Teacher (NTA)	1	15	0
4.	ANS Science Teachers Workshop	1	34	0
5.	Anchor School at Sidney Lanier	1	2	10
6.	ATHENA Middle School Girls Science Workshop	1	1	7
7.	Belleview/Fairview/Swift Creek Middle Schools *(COE Minority Outreach)*	1	5	46
8.	Boone High School - Orlando	1	1	1
9.	Brevard Community College	1	1	1
10.	Career Shadowing Day	2	1	31
11.	Carol City High School *(COE Minority Outreach)*	1	3	20
12.	College of Engineering Recruiting Days (High School Students)	2	2	47
13.	Conestoga High School	1	2	1
14.	Conway High School *(COE Minority Outreach)*	1	5	26
15.	CPET Science, Engineering & Humanities Symposium	2	8	16
16.	Crystal River High School	1	2	1
17.	Cypress Creek High School *(COE Outreach)*	1	1	21
18.	Discovery Intermediate School (Honors Physics)	3	6	25
19.	Eye on Engineering Summer Camp (High School)	1	2	18
20.	Francis Marion University	1	1	2
21.	F.W. Springstead High School	5	3	4
22.	Gainesville Country Day School	1	5	18
23.	GATORTRAX Middle School Honors Workshop	2	1	28
24.	George Washington University	1	1	1
25.	Graduate Student Recruiting Weekend	1	1	16
26.	Hampton Roads Academy	4	2	1
27.	Hillsborough Community College	3	1	27
28.	Howard Bishop Middle School	2	2	55
29.	HPS Student Section Science Teachers Workshop	1	9	2
30.	Johnson Middle School	1	3	1
31.	Kishneva Academy for Boys - Alachua	1	1	4
32.	Lecanto High School	23	2	7
33.	Lincoln Middle School *(COE Minority Outreach)*	3	5	63
34.	Middleton High School - Tampa *(COE Minority Outreach)*	1	3	56
35.	National City Teacher's Workshop	1	47	0

(Table III-1A continues on next page.)

TABLE III-1A

REACTOR SHARING PROGRAM
SUMMARY OF USAGE OF UFTR FACILITIES
(September 2004 – August 2005)

	School	Usages*	Faculty	Students
36.	North Bethesda Teacher's Workshop	1	25	0
37.	Ohio State University	1	1	1
38.	Orlando Memorial Middle School *(COE Outreach)*	2	3	35
39.	Outstanding High School Scholars Program	1	2	13
40.	Palmer Trinity School	1	2	1
41.	PEEK Middle School Boys Science Workshop	1	1	26
42.	P.K. Yonge High School (Honors Physics)	2	1	32
43.	Ridgeview High School IB Program	3	2	6
44.	Robert F. Munroe High School	2	2	1
45.	Saint Patrick Middle School	5	7	74
46.	Sandalwood High School	1	1	1
47.	Santa Fe Community College	9	5	52
48.	Savannah State University	2	10	54
49.	South Carolina State University	2	4	12
50.	Student Science Training Program (Summer Research) (High Schools)	9	3	96
51.	Union County High School	10	1	3
52.	University of Central Florida	48	4	2
53.	University of Memphis	1	3	0
54.	University of San Francisco	1	1	0
55.	University of Southern California	2	1	2
56.	University External Facility Visitors/Student Communications	2	11	1
57.	Valencia Community College	1	1	1
58.	West Orange High School *(COE Outreach)*	1	3	25
	TOTAL	**185**	**276**	**1047**

*Usage is defined as utilization of the University of Florida Training Reactor facilities for all or any part of a day with the average being over four hours. In many cases, a school can have multiple usages but all related to the same research project such as two projects for Lecanto High School that involved long term irradiations as did others such as one project for Ridgeview High School IB Program, one for Robert F. Munroe High School and especially the one project for researchers at the University of Central Florida Physics Department.

TABLE III-1B

REACTOR SHARING PROGRAM
SUMMARY OF SELECTED FACILITY UTILIZATION
(September 2004 – August 2005)

NOTE: The projects marked with one asterisk (*) indicate irradiations or neutron activations. The projects marked with two asterisks (**) indicate training/ educational use. The projects marked with three asterisks (***) indicate demonstrations of reactor operations and other uses. "Experiment Time" is total time that the facility dedicates to a particular use; it includes "Run Time." "Run Time" is inclusive time commencing with reactor startup and ending with shutdown and securing of the reactor.

Project and User	Type of Activity	Run Time Hours	Experiment Time Hours
*Center for Precollegiate Education and Training – NAA Research on Effects of Hard Versus Soft Water on Prevalence of Heart Disease – M.S. Knight and S. Walker, Robert F. Munroe High School /Dr. W.G. Vernetson, UF – Reactor Sharing	Summer 2004 Student Research Program Project – Evaluation and Quantification of Trace Element Content in Various Hard Versus Soft Water Evaporative Samples for Correlation with Incidence of Heart Disease for Student Ben Stewart of Robert F. Munroe High School (Local and Regional Science Fair Winner and Junior Science, Engineering and Humanities Symposium Participant) (Continued)	0.00	1.42
*Center for Precollegiate Education and Training – NAA Research on Effects of Gatorade Consumption on Trace Element Composition of Hair – Ms. Janis Tobin and Ms. G.M. Keyes, Palmer Trinity School / Dr. W.G. Vernetson, UF – Reactor Sharing	Summer 2004 Student Research Program Project – Evaluation and Quantification of Variable Trace Element Metal Content of Various Hair Samples Dependent Upon Gatorade Consumption for Student Garrett deRosset of Palmer Trinity School (Local Science Fair Entrant and Junior Science, Engineering and Humanities Symposium Participant) (Continued)	0.00	2.25
***Familiarization Tour for George Washington University Physics Student – Dr. W.G. Vernetson, UF – Reactor Sharing	Detailed Walk-through Tour of Reactor and NAA Laboratory to Discuss Usage, Capabilities and Operations Including Curricular and Research Opportunities for Potential Nuclear Engineering Graduate Student Sarmadi Almecci	0.00	1.00

TABLE III-1B

REACTOR SHARING PROGRAM
SUMMARY OF SELECTED FACILITY UTILIZATION
(September 2004 – August 2005)

Project and User	Type of Activity	Run Time Hours	Experiment Time Hours
***Familiarization Tour for Support for Career Shadowing Day – Dr. W.G. Vernetson, UF – Reactor Sharing	Detailed Walk-through Tours of Reactor and NAA Laboratory to Discuss Usage, Capabilities and Operations Including Curricular Use and Career-related Opportunities for COE Mentors Matt Lambie and Sarah Granger with Two Groups of High School Seniors to Provide Insight into Careers Involving Nuclear and Radiological Engineering	0.00	2.67 (0.67)
**Discovery Intermediate School Physical Science Honors Students – CPET Science Coordinator J. Bokor / Mary Pollock, Discovery Intermediate School and W.G. Vernetson, UF – Reactor Sharing	Lectures, Tours and Demonstrations of UFTR Operations with Radiation Surveys and Exercises to Measure Half-life of Irradiated Elements and in Using the Rabbit System and PC-based Analyzers for Trace Element Analysis of Hair Samples Using NAA Techniques Plus Contamination Control Exercises Using Anticontamination Clothing with Subsequent Trace Element Analysis of Series of Hair Samples for Physical Science Honors Students	3.78 (0.50)	8.75 (0.58)
***High School Senior Outreach for Recruitment to Engineering / Nuclear Engineering – Ms. Deb Mayhew and Ms. Yolanda Hankerson (COE) – Reactor Sharing	Series of Lectures and Walk-through Tours of Reactor and NAA Laboratory Facilities Including Use of Survey Meters and Demonstration of Trace Element and Other Analytical Capabilities for High School Students and Parents Interested in Nuclear and Radiological Engineering and/or Other Engineering Areas	0.42 (0.42)	2.17 (1.17)
**Memorial Middle School, Orlando – Mr. Earl Wade (COE) / Dr. W.G. Vernetson, UF – Reactor Sharing	Lecture, Tour and Demonstration of Reactor and NAA Laboratory Operations Including Radiation Surveys of Everyday Objects and Use of the Rabbit system and PC-based Analyzers for Memorial Middle School Science Students and Teachers as Part of Minority Outreach Program	0.00	3.83
*NAA Research to Identify Trace Element Variations in Fish Depending Upon Living Zone – Mr. Ron Worthington, Lecanto High School / Dr. W.G. Vernetson, UF – Reactor Sharing	NAA Evaluation of Trace Heavy Elements in Different Species of Fresh Gulf of Mexico Fish Dependent Upon the Living Zone from the Gulf Surface to the Gulf Floor for a Science Fair Project for Student Sneha Patel of Lecanto High School (Local/Regional Winner/Placed Third at State Science Fair)	21.25 (0.08)	43.25 (3.00)

TABLE III-1B

REACTOR SHARING PROGRAM
SUMMARY OF SELECTED FACILITY UTILIZATION
(September 2004 – August 2005)

Project and User	Type of Activity	Run Time Hours	Experiment Time Hours
*NAA Research to Identify Trace Element Variations in Wild Versus Farm-Raised Salmon - Mr. Ron Worthington, Lecanto High School / Dr. W.G. Vernetson, UF – Reactor Sharing	NAA Evaluation of Trace Heavy Element Variations in Different Species of Wild Versus Farm-Raised Salmon for a Science Fair Project for Student Lakshmi Ram of Lecanto High School (Local/Regional Winner/Placed Second at State Science Fair)	20.23 (0.25)	53.92 (1.92)
*NAA Research to Identify Trace Element Variations in Fresh Citrus Dependent Upon Organic Versus Non-organic Groves – Ms. Renae Allen, Union County High School / Dr. W.G. Vernetson, UF – Reactor Sharing	NAA Evaluation of Trace Element Variations in Fresh Citrus to Quantify Effects of Soil Treatment (Pesticides/Herbicides) on Fruit from Non-organic Versus Organic Groves for a Science Fair Project for Student Holly Dellenger of Union County High School (Local/Regional Winner/State Science Fair Participant)	14.67 (0.08)	33.58 (1.00)
**Anchor School at Sidney Lanier, Gainesville – Dr. W.G. Vernetson, UF – Reactor Sharing	Lecture, Tour and Demonstration of Reactor and NAA Laboratory Operations Including Radiation Surveys of Everyday Objects and Use of the Rabbit System and PC-based Analyzers for Anchor School at Sidney Lanier Students and Teachers as Part of Engineering Fair Activities	0.00	2.33 (0.67)
**Lincoln Middle School, Gainesville – Mrs. Carla Garcia, LMS / Dr. W.G. Vernetson, UF – Reactor Sharing	Lecture, Tour and Demonstration of Reactor and NAA Laboratory Operations Including Radiation Surveys of Everyday Objects and Use of the Rabbit System and PC-based Analyzers for Lincoln Middle School Science Students and Teachers as Part of Engineering Fair Activities	0.00	2.58 (0.25)
**Kishneva Academy for Boys, Alachua – Mrs. K. Bellach, KAB/Dr. W.G. Vernetson, UF – Reactor Sharing	Lecture, Tour and Demonstration of Reactor and NAA Laboratory Operations Including Radiation Surveys of Everyday Objects and Use of the Rabbit System and PC-based Analyzers for Kishneva Academy for Boys Science Students and Teachers as Part of Engineering Fair Activities	0.00	2.58 (2.58)
Administrative and Education Communication Activities – Dr. W.G. Vernetson, UF – Reactor Sharing	Scheduling of Future Year Usages and Communications of Power and Non-power Reactor Usage and Capabilities and Operations Information to Support Academic Efforts at Various Schools Plus Reporting and Communications Activities and Several Student Visits for Non-specific Schools	0.00	35.17 (1.08)

TABLE III-1B

REACTOR SHARING PROGRAM
SUMMARY OF SELECTED FACILITY UTILIZATION
(September 2004 – August 2005)

Project and User	Type of Activity	Run Time Hours	Experiment Time Hours
**Swift Creek/ Fairview/ Belleview Middle Schools, Orlando – Dr. Jonathan Earle (COE) / Dr. W.G. Vernetson, UF – Reactor Sharing	Lecture, Tour and Demonstration of Reactor and NAA Laboratory Operations Including Radiation Surveys of Everyday Objects and Use of the Rabbit System and PC-based Analyzers for Swift Creek, Fairview and Belleview Middle School Honors Science Students and Teachers as Part of Minority Outreach Program	0.50	4.42 (0.58)
**J.K. Davis Conference Center – Mr. Chuck Vincent (ANS)/ Dr. W.G. Vernetson, UF – Reactor Sharing	Lectures and Demonstrations as Part of High School Teacher Workshops Including Radiation Surveys of Everyday Objects and Utilization and Applications of UFTR Reactor and NAA Laboratory Facilities	0.00	12.58 (0.08)
***Demonstration of Reactor and NAA Laboratory Operations for Educational Applications – Ms. Kristin Russell, Crystal River Charter High School / Dr. W.G. Vernetson, UF – Reactor Sharing	Series of Lectures, Tours and Demonstrations of UFTR and NAA Laboratory Operations with Discussion of Facility Usage and Capabilities for Education and Training Including Simulated Measurement of Half-Life of Radionuclides and Trace Element Analysis of Hair Samples Plus Contamination Control Exercises for Crystal River Charter High School Academy of Environmental Science Students	1.98	4.08
***Engineering Fair Outreach and Support – Dr. W.G. Vernetson – Reactor Sharing	Efforts to Support ANS Student Section Public Education Efforts for Annual Engineering Fair and Fall Student Society Fair	0.00	1.00 (1.00)
**Carol City High School – Dr. Jonathan Earle (COE) / Dr. W.G. Vernetson, UF – Reactor Sharing	Lecture, Tour and Demonstration of Reactor and NAA Laboratory Operations Including Radiation Surveys of Everyday Objects and Use of the Rabbit System and PC-based Analyzers for Carol City High School Honors Science Students and Teachers as Part of Minority Outreach Program	0.00	2.00
**Conway Middle School, Orlando – Dr. Jonathan Earle (COE) / Dr. W.G. Vernetson, UF – Reactor Sharing	Lecture, Tour and Demonstration of Reactor and NAA Laboratory Operations Including Radiation Surveys of Everyday Objects and Use of the Rabbit System and PC-based Analyzers for Conway Middle School Honors Science Students and Teachers as Part of Minority Outreach Program	0.00	1.75
**Familiarization Tour for Brain Institute Representatives – Dr. Albina Mikhaphova (UFBI)/ Dr. W.G. Vernetson, UF – Reactor Sharing	Walk-through Tour of Reactor and NAA Laboratory Facilities to Discuss Usage and Capabilities for Trace Element Analysis of Biological Samples for Tissue Comparisons for Various Interested Brain Institute Research Associates	0.00	1.17

TABLE III-1B

REACTOR SHARING PROGRAM
SUMMARY OF SELECTED FACILITY UTILIZATION
(September 2004 – August 2005)

Project and User	Type of Activity	Run Time Hours	Experiment Time Hours
**P.K. Yonge High School Honors Physics Program – Dr. Griff Jones, PKYHS/Dr. W.G. Vernetson, UF – Reactor Sharing	Series of Lectures, Tours and Demonstrations of UFTR Operations and Comparison with Power Reactors with Radiation Surveys and NAA Training Exercises Demonstrating Isotope Identification, Half-life Measurement and Trace Element Analysis of Previously Irradiated Hair Samples Using the Rabbit System PC-based Analyzers	0.00	7.00
**Gainesville Country Day School Science Classes – Mr. Kelly Childers, GCDS – Reactor Sharing	Lectures, Tours and Demonstrations of UFTR Operations with Radiation Surveys and Exercises to Measure Half-life of Irradiated Elements and in Using the Rabbit System and PC-based Analyzers for Trace Element Analysis of Hair Samples Using NAA Techniques Plus Contamination Control Exercises Using Anticontamination Clothing with Subsequent Trace Element Analysis of Series of Hair Samples	0.80	5.58
***Center for Precollegiate Education 42nd Annual Junior Science, Engineering and Humanities Symposium – Dr. MaryJo Koroly and Ms. Deborah Paulin (CPET) – Reactor Sharing	Series of Lectures, Tours and Demonstrations of Reactor and NAA Laboratory Facility Operations, Capabilities and Applications for Honors Groups of High School Junior/Senior Level Students and Teachers Including Various Support Activities	0.00	7.08
**Lincoln Middle School, Gainesville – Dr. Jonathan Earle (COE) / Dr. W.G. Vernetson, UF – Reactor Sharing	Lectures, Tours and Demonstrations of Reactor and NAA Laboratory Operations Including Radiation Surveys of Everyday Objects and Use of the Rabbit System and PC-based Analyzers for Lincoln Middle School Honors Science Students and Teachers as Part of Minority Outreach Program	0.00	4.00
**Howard Bishop Middle School, Gainesville – Dr. Jonathan Earle (COE) / Dr. W.G. Vernetson, UF – Reactor Sharing	Lectures, Tours and Demonstrations of Reactor and NAA Laboratory Operations Including Radiation Surveys of Everyday Objects and Use of the Rabbit System and PC-based Analyzers for Lincoln Middle School Honors Science Students and Teachers as Part of Minority Outreach Program	0.00	3.50
**Familiarization Visit for Alachua County Science Teachers – Dr. Skip Ingley/Dr. W.G. Vernetson, UF – Reactor Sharing	Lecture and Walk-through Tour of Reactor and NAA Laboratory Facilities to Discuss Usage and Capabilities for Alachua County Precollegiate Science Teachers to Understand Possible Support of Curriculum Topics	0.00	1.58

TABLE III-1B

REACTOR SHARING PROGRAM
SUMMARY OF SELECTED FACILITY UTILIZATION
(September 2004 – August 2005)

Project and User	Type of Activity	Run Time Hours	Experiment Time Hours
***Demonstration of Reactor and NAA Laboratory Operations for Educational Applications – Ms. Esther Branch, St. Patrick School / Julie Bokor, CPET / Dr. W.G. Vernetson, UF – Reactor Sharing	Lectures, Tours and Demonstrations of UFTR and NAA Laboratory Operations with Discussion of Facility Usage and Capabilities for Education and Training Including Measurement of Half-Life of Radionuclides and Trace Element Analysis of Hair Sample Irradiated Via Rabbit System for St. Patrick School Science Students	10.30 (1.62)	18.33 (1.83)
**Ridgeview High School AP Physics Class – Mr. Robert Lowery, RHS – Reactor Sharing	Lectures, Tours and Demonstrations of UFTR Operations with Radiation Surveys and Exercises to Measure Half-life of Irradiated Elements and in Using the Rabbit System and PC-based Analyzers for Trace Element Analysis of Hair Samples Using NAA Techniques Plus Contamination Control Exercises Using Anticontamination Clothing with Subsequent Planned Trace Element Analysis of Series of Hair Samples	1.10	6.83
*South Carolina State University Nuclear Engineering Department – Dr. Tica Valdes and Ms. April Hutton, SCSU/ Dr. W.G. Vernetson, UF – Reactor Sharing	Series of Lectures, Tours and Demonstrations of UFTR Operations and Comparison with Power Reactors with Radiation Surveys and NAA Training Exercises Demonstrating Isotope Identification, Half-life Measurement and Trace Element Analysis of Hair Samples Using the Rabbit System PC-based Analyzers Plus Contamination Surveys Plus Collection of Hair Samples for Follow-up Trace Element Analysis for Nuclear Engineering Faculty and Students Plus Selected Committed High School Students	1.10	13.42 (1.75)
**Santa Fe Community College Nuclear Medicine Technology Program – Mr. Karl Eckberg and Ms. Angela Conti, SFCC – Reactor Sharing	Lecture, Tour and Demonstration of UFTR Operations with Radiation Surveys and NAA Training Exercises Demonstrating Isotope Identification, Half-life Measurement and Trace Element Analysis of Hair Samples Using the Rabbit System PC-based Analyzers Plus Demonstration of Gas Flow Proportional Counter for Contamination Surveys Plus Follow-up Trace Element Analysis of Hair Samples	2.43	6.75

TABLE III-1B

REACTOR SHARING PROGRAM
SUMMARY OF SELECTED FACILITY UTILIZATION
(September 2004 – August 2005)

Project and User	Type of Activity	Run Time Hours	Experiment Time Hours
*NAA Research to Quantify Trace Element Content of Various Potable Water Sources for IB Project – Mr. Robert Lowery/Dr. Barbara Gruber, Ridgeview High School / Dr. W.G. Vernetson, UF – Reactor Sharing	International Baccalaureate Degree Research Project to Quantify and Compare Variations in Trace Metal Content of Various Potable Water Supplies for IB Student Jessica Schwab and Her Student Team at Ridgeview High School	2.10	7.25 (0.17)
**Familiarization Tour for Potential NRE Graduate Students – Dr. W.E. Bolch / Dr. W.G. Vernetson, UF – Reactor Sharing	Detailed Walk-through Tour of Reactor and NAA Laboratory Facilities to Discuss Usage Capabilities and Operations Plus Curricular Usage for Students from Various Schools (Michigan, Tennessee, Missouri – Rolla, Auburn, Francis Marion, etc.) as Potential NRE Graduate Students	0.00	1.25
**Cypress Creek High School (Orlando) Honors Science Students – Mr. Robert Connor, CCHS and D. Mayhew/W.G. Vernetson, UF – Reactor Sharing	Lecture, Tour and Demonstration of UFTR Operations with Radiation Surveys and Exercises to Simulate Measurement of Half-life of Irradiated Elements and in Using the Rabbit System and PC-based Analyzers for Trace Element Analysis of Previously Irradiated Hair Samples Using NAA Techniques for Cypress Creek High School Honors Science Students as Part of COE Outreach Effort	0.00	2.67
**West Orange High School (Orlando) Honors Science Students – Mrs. Martha Patterson, WOHS and D. Mayhew/W.G. Vernetson, UF – Reactor Sharing	Lecture, Tour and Demonstration of UFTR Operations with Radiation Surveys and Exercises to Simulate Measurement of Half-life of Irradiated Elements and in Using the Rabbit System and PC-based Analyzers for Trace Element Analysis of Previously Irradiated Hair Samples Using NAA Techniques for West Orange High School Honors Science Students as Part of COE Outreach Effort	0.00	1.83
***Familiarization Tour for Conestoga High School Student – Dr. W.G. Vernetson, UF – Reactor Sharing	Detailed Walk-through Tour of Reactor and NAA Laboratory to Discuss Usage, Capabilities and Operations Including Curricular Use for Potential Nuclear Engineering Student Thomas Robinson and His Parents	0.00	1.25

TABLE III-1B

REACTOR SHARING PROGRAM
SUMMARY OF SELECTED FACILITY UTILIZATION
(September 2004 – August 2005)

Project and User	Type of Activity	Run Time Hours	Experiment Time Hours
**Santa Fe Community College Medical Radiography Program – Ms. Bobbie Konter and Mr. Karl Eckberg, SFCC – Reactor Sharing	Lectures, Tours and Demonstrations of UFTR Operations with Radiation Surveys and NAA Training Exercises Demonstrating Isotope Identification and Trace Element Analysis Technique on Hair Samples Using the Rabbit System PC-based Analyzers Plus Demonstration of Gas Flow Proportional Counter for Contamination Surveys Plus Follow-up Trace Element Analysis of Hair Samples	5.97 (1.88)	13.58 (2.00)
**HPS Precollegiate Teacher Workshop – Dr. W.E. Bolch / Dr. W.G. Vernetson, UF – Reactor Sharing	Lecture, Tour and Demonstration of UFTR Operations with Radiation Surveys and NAA Training Exercises Demonstrating Isotope Identification, Half-life Measurement and Trace Element Analysis of Hair Samples Using the Rabbit System PC-based Analyzers for a Group of Precollegiate Science Teachers from Around North Central Florida as Part of a Workshop Organized by the UF Student Health Physics and Medical Physics Societies	0.77	5.50
**Hillsborough Community College Nuclear Medicine and Radiation Therapy Technology Program – Dr. Larry Gibson, HCC – Reactor Sharing	Lecture, Tour and Demonstration of Facility Operations with Radiation Surveys and Exercise in Use of Rabbit System for Activation for Half-life Measurements and Trace Element Analysis of Hair Samples Using NAA Techniques and Demonstration of Neutron Radioisotope Production and Use of Gas Flow Proportional Counters	6.17	12.50
**Boone High School, Orlando/ Dr. W.G. Vernetson, UF – Reactor Sharing	Detailed Walk-through Tour and Demonstration of Reactor and NAA Laboratory Operations Including Radiation Surveys of Everyday Objects and Use of the Rabbit System and PC-based Analyzers for Boone High School Honors Science Student and Parents	0.00	2.83 (0.67)
***Familiarization Tour for Sandalwood High School Student – Dr. W.G. Vernetson, UF – Reactor Sharing	Detailed Walk-through Tour of Reactor and NAA Lab to Discuss Usage, Capabilities and Operations Including Curricular Use for Potential Nuclear Engineering Honors Student Cheryl Clark of Sandalwood High School and Her Mother	0.00	0.75
***Familiarization Tour for Francis Marion University Physics Students – Dr. W.G. Vernetson, UF – Reactor Sharing	Detailed Walk-through Tour of Reactor and NAA Lab to Discuss Usage, Capabilities and Operations Including Curricular Use for Francis Marion University Physics Students as Potential Enrollment as Graduate Students in Nuclear and Radiological Engineering	0.00	1.17

TABLE III-1B

REACTOR SHARING PROGRAM
SUMMARY OF SELECTED FACILITY UTILIZATION
(September 2004 – August 2005)

Project and User	Type of Activity	Run Time Hours	Experiment Time Hours
***Familiarization Tour for Ohio State University Engineering-Physics Student – Dr. W.G. Vernetson, UF – Reactor Sharing	Detailed Walk-through Tour of Reactor and NAA Laboratory to Discuss Usage, Capabilities and Operations Including Curricular Use for Engineering-Physics Student Ryanne Kennedy from Ohio State University as a Potential NRE Graduate Student	0.00	1.25
*Basic Physics Research to Support Transmutation Studies for Solid State Laser Development – Dr. R.F. Peale and Dr. E. Flitsiyan, University of Central Florida / Dr. W.G. Vernetson, UF – Reactor Sharing	Irradiation of Isotopically Pure Zn-68 Foils for Transmutation Studies to Evaluate Energy Level Changes Support Basic Physics Research for Zinc-based Solid State Laser Development	127.22 (64.17)	145.17 (70.25)
**North Bethesda Middle School – Mr. Chuck Vincent (ANS)/ Dr. W.G. Vernetson, UF – Reactor Sharing	Lectures and Demonstrations as Part of Precollegiate Teacher Workshops Including Radiation Surveys of Everyday Objects and Utilization and Applications of UFTR Reactor and NAA Laboratory Facilities	0.00	14.00 (2.00)
***Familiarization Tour for Brevard Community College Pre-engineering Student – Dr. W.G. Vernetson, UF – Reactor Sharing	Detailed Walk-through Tour of Reactor and NAA Laboratory to Discuss Usage, Capabilities and Operations Including Opportunities for Potential Undergraduate Assistant Work for Brevard Community College Student Crystal McJunkin	2.12 (1.62)	2.17 (1.67)
***Center for Precollegiate Education and Training – Dr. MaryJo Koroly and Ms. Deborah Paulin (CPET)/ Dr. W.G. Vernetson, UF – Reactor Sharing	Lecture and Demonstrations on Reactor Operations and Usage Comparing UFTR with Power Reactors for Assembled Summer Science Research Training Program Participants (High School Students) and Non-UF College Student Mentors with Subsequent Facility Tours for a Number of Participants	0.00	5.33
***Familiarization Tours for Visiting University / Other Faculty / Industry Instructors – Dr. W.G. Vernetson, UF – Reactor Sharing	Series of Walk-through Tours of Reactor and NAA Laboratory Facilities to Discuss Capabilities, Usage and Operations Along with Nuclear Engineering Education Opportunities for Various Outside University Faculty and Other Visitors and Industry Instructors Plus Accompanying Students	0.00	2.42

TABLE III-1B

REACTOR SHARING PROGRAM
SUMMARY OF SELECTED FACILITY UTILIZATION
(September 2004 – August 2005)

Project and User	Type of Activity	Run Time Hours	Experiment Time Hours
**National City School Board Education Center – Mr. Chuck Vincent (ANS)/Dr. W.G. Vernetson, UF – Reactor Sharing	Lectures and Demonstrations as Part of High School Teacher Workshops Including Radiation Surveys of Everyday Objects and Utilization and Applications of UFTR Reactor and NAA Laboratory Facilities	0.00	11.50
***TREAT Workshops – Dr. Kenneth Sajwan, Savannah State University/Dr. W.G. Vernetson, UF – Reactor Sharing	Series of Lectures and Demonstrations Comparing Nonpower UFTR to Power Reactors and Technology Applications as Part of Teaching Radiation, Energy and Technology (TREAT) Workshop for Savannah State University Teachers, Students, High School Teachers and Community Members	0.00	21.92 (0.33)
***PEEK Middle School Boys Workshop – J. Citty (COE) / Dr. W.G. Vernetson, UF – Reactor Sharing	Lecture, Tour and Demonstration of Reactor and NAA Laboratory Facility Operations Including Use of Survey Meters and Source Location Exercise Plus Demonstration of Half-life Measurement and Trace Element Analysis of Irradiated Hair Samples for Middle School Boys PEEK Workshop Group	0.75 (0.33)	3.58 (1.50)
***ATHENA Middle School Girls Workshop – Karen Bray (COE) / Dr. W.G. Vernetson, UF – Reactor Sharing	Lectures, Tours and Demonstrations of Reactor and NAA Laboratory Facility Operations Including Use of Survey Meters and Source Location Exercise Plus Demonstration of Half-life Measurement and Trace Element Analysis of Irradiated Hair Samples for Middle School Girls ATHENA Workshop Group	0.75 (0.42)	3.17 (1.33)
***Eye on Engineering Summer Camp Workshop – J. Brady (COE) / Dr. W.G. Vernetson, UF – Reactor Sharing	Lectures, Tours and Demonstrations of Reactor and NAA Laboratory Facility Operations Including Use of Survey Meters and Source Location Exercise Plus Half-life Measurement and Trace Element Analysis of Irradiated Hair Samples for Upper Level High School Students in Eye on Engineering Summer Camp	0.93	3.83
**Student Science Training Program for High School Student Researchers – Dr. M.J. Koroly / D. Paulin, CPET / Dr. W.G. Vernetson, UF – Reactor Sharing	Series of Lectures, Tours and Demonstrations of Facility Capabilities and Operations Including Various Hands-on Instruction, Half-life Measurements, Trace Element Analysis and Other Activity Participation for Two SSTP High School Juniors to Allow Selection and Preparation for Performing Summer Research Projects (Mike Ferebee, Hampton Roads Academy and David Roebuck, F.W. Springstead HS)	4.97 (1.50)	29.00 (8.17)

TABLE III-1B

REACTOR SHARING PROGRAM
SUMMARY OF SELECTED FACILITY UTILIZATION
(September 2004 – August 2005)

Project and User	Type of Activity	Run Time Hours	Experiment Time Hours
***GATORTRAX Middle School Honors Workshop – J. Brady (COE) / Dr. W.G. Vernetson, UF – Reactor Sharing	Lecture, Tour and Demonstration of Reactor and NAA Laboratory Facility Operations Including Use of Survey Meters and Source Location Exercise Plus Half-life Measurement and Trace Element Analysis of Hair Samples for Middle School Workshop Groups of Honors Science Students	2.88	7.33
**Middleton High School, Tampa – Ms. Yolanda Hankerson (COE)/Dr. W.G. Vernetson, UF – Reactor Sharing	Lectures, Tours and Demonstrations of Reactor and NAA Laboratory Operations Including Radiation Surveys of Everyday Objects and Use of the Rabbit System and PC-based Analyzers for Middleton High School Science Honors Students and Teachers as Part of COE Outreach Program	0.00	3.33
**Florida High School Merit Scholars Program – Mr. Jason Thomas, Admissions/ Dr. W.G. Vernetson, UF – Reactor Sharing	Lecture and Tour for Outstanding High School Student Merit Scholars Program Including Students and Parents to Discuss Facility Usage and Capabilities to Attract Superior Students into Nuclear and Radiological Engineering	0.00	1.33
***Center for Precollegiate Education and Training Science Quest Middle School Student Workshop – Ms. Julie Bokor (CPET), Ms. Allison Baird, Alachua County Teacher/ Dr. W.G. Vernetson, UF – Reactor Sharing	Series of Lectures, Tour and Demonstrations of Reactor and NAA Laboratory Operations Including Radiation Surveys of Everyday Objects, Measurement of Half-life, Demonstration Use of the Rabbit System and PC-based Analyzers to Determine Trace Element Content of Previously Irradiated Hair Samples Plus Contamination Control Exercises Involving Dress Out in Anticontamination Clothing and Use of Robots for Demonstration Purposes for Two Workshops	0.00	8.25
**Santa Fe Community College Fundamentals of Physical Science, PSC-1341 – Dr. Vince Bourke, SFCC – Reactor Sharing	Series of Lectures, Tours and Demonstrations of UFTR Operations with Radiation Surveys and NAA Training Exercises Demonstrating Isotope Identification, Half-life Measurement and Trace Element Analysis of Hair Samples Using the Rabbit System PC-based Analyzers Plus Collection of Hair Samples for Follow-up Analysis	0.67	7.00
**Santa Fe Community College General Physics, PHY-2005 – Dr. Vince Bourke, SFCC – Reactor Sharing	Lecture, Tour and Demonstration of UFTR Operations with Radiation Surveys and NAA Training Exercises Demonstrating Isotope Identification, Half-life Measurement and Trace Element Analysis of Hair Samples Using the Rabbit System PC-based Analyzers Plus Collection of Hair Samples for Follow-up Analysis	0.70	3.08

TABLE III-1B

REACTOR SHARING PROGRAM
SUMMARY OF SELECTED FACILITY UTILIZATION
(September 2004 – August 2005)

Project and User	Type of Activity	Run Time Hours	Experiment Time Hours
*Center for Precollegiate Education and Training – NAA Research on Trace Element Composition of Sediments Around Florida Springs – Mr. C.L. Wagner and Mrs. J. Eaton, F.W. Springstead High School/ Dr. W.G. Vernetson, UF – Reactor Sharing	Summer 2005 Student Research Program Project (Continued) – Evaluation and Quantification of Variable Trace Element Content of Sediments Around the Weeki-Wachee Spring Drainage Area for Student David Roebuck of F.W. Springstead High School	0.00	18.33 (6.42)
*Center for Precollegiate Education and Training – NAA Research on Variation in Trace Metal Composition of Ground Coffee – Dr. R.A. Williams and Mr. S. Wells, Hampton Roads Academy/Dr. W.G. Vernetson, UF – Reactor Sharing	Summer 2005 Student Research Program Project – Evaluation and Quantification of Variable Trace Metal Content of Various Ground Coffees Dependent on Grinding Methodology for Student Mike Ferebee of Hampton Roads Academy	0.00	10.67 (4.92)
	TOTAL	**234.56 (72.87)**	**649.81 (117.59)**

1. Values in parentheses represent multiple or concurrent facility utilization (run or experiment time); that is, the reactor was already being utilized in a primary run or activity for a project so a reactor training or demonstration utilization could be conducted concurrently with a scheduled NAA irradiation, course experiment, or other reactor run.

2. Experiment time is run time (total key on time minus checkout time) plus set-up time for experiments or other reactor or facility usage.

3. These hours do not reflect the hundreds of hours of NAA Laboratory usage for analysis of irradiated samples, only a small part of which is charged to the Reactor Sharing Grant.

TABLE III-2

MONTHLY REACTOR ENERGY GENERATION [1]
(September 2004 – August 2005)

Month	Energy Generation Monthly Ranking [2]	KW-Hrs	Hours at Full Power
September 2004	10	774.140	7.633
October 2004	9	1,291.673	12.680
November 2004	3	2,504.000	24.784
December 2004	2	2,926.438	29.299
January 2005	4	2,095.218	20.548
February 2005	6	1,853.430	18.082
March 2005	1	3,549.595	31.750
April 2005	8	1,339.003	13.150
May 2005	5	2,047.765	20.150
June 2005	7	1,470.626	13.682
July 2005	11	47.333	0.450
August 2005	12	1.809	0.000
YEARLY TOTAL		**19,901.030**[3]	**192.208**

[1] The yearly total energy generation of 19.901 megawatt-hours for the 2004–5 reporting year represents a 36.91% increase from last year's total of 14.536 megawatt-hours, while the 192.208 hours at full power represents a 38.98% increase from the previous yearly total of 138.304 hours. With no large outages for most of the year and one part time SRO until March 22, 2005, plus the Facility Director, and a second RO not certified until near year's end on August 22, 2005, operator unavailability was a significant contributor to reactor unavailability and the relatively low energy generation for the year. For the 2004–5 reporting year, the energy generation is higher essentially due to the high availability as forced unavailability was at 49½ days with no forced outages lasting more than 4 days until July and August 2005.

[2] This column showing the ranking of monthly energy generation is included for potential correlation with results of environmental monitoring in Chapter VII, though such correlations have not been seen in the past and were not seen in this year's data as well.

[3] The 19,901.030 kilowatt-hours energy generation for the 2004–5 year ranks third in the past ten-year period. This relatively high ranking is due to less forced outage time during this reporting year.

TABLE III-3

MONTHLY REACTOR USAGE/AVAILABILITY DATA
(September 2004 – August 2005)

Month	Key-On Time	Exp. Time[1]	Run Time[2]	Availability[3]
September 2004	21.20 hrs.	231.00 hrs.	19.03 hrs.	69.17%
October 2004	21.60 hrs.	319.58 hrs.	18.68 hrs.	95.97%
November 2004	34.60 hrs.	235.92 hrs.	32.38 hrs.	92.08%
December 2004	49.10 hrs.	227.67 hrs.	45.58 hrs.	85.89%
January 2005	34.80 hrs.	251.17 hrs.	32.42 hrs.	92.74%
February 2005	34.00 hrs.	243.83 hrs.	31.27 hrs.	99.55%
March 2005	52.50 hrs.	275.08 hrs.	48.30 hrs.	64.11%
April 2005	21.30 hrs.	221.42 hrs.	18.23 hrs.	99.17%
May 2005	40.30 hrs.	316.17 hrs.	35.35 hrs.	93.15%
June 2005	34.00 hrs.	282.92 hrs.	29.60 hrs.	79.17%
July 2005	6.70 hrs.	280.83 hrs.	1.52 hrs.	12.50%
August 2005	5.50 hrs.	340.25 hrs.	1.45 hrs.	5.65%
YEARLY TOTAL	**355.60 hrs.**	**3,225.84 hrs.**	**313.81 hrs.**	**74.10%**

[1] Experiment time is run time (total key-on time minus checkout time) plus set-up time for experiments, tours, or other facility usage including checkouts, tests and maintenance involving reactor running or facility usage.

[2] The three categories of facility usage data in this table show relatively small but significant increases over the previous year, especially those related to reactor operations. Key-on time is up 17.36% while run time is up 15.31%, limited primarily by availability of reactor operators. With only two operators including one working about 75% time and no second operator from March 23 through August 21, 2005, personnel availability continued to be poor as efforts to hire part time operators were not very successful. Experiment time, as well, is increased slightly by 2.96% showing a continued emphasis for class usage as the experiment time was well used for research, training and education during this past year, especially related to reactor sharing visiting groups and one large research irradiation but also a growing number of on-campus groups plus better accounting of facility-related activities.

[3] Average availability on a yearly basis is 74.10% as shown above and 73.77% per Table III-4. As in recent years, this availability accounts for lost availability for administrative reasons as well as for repair and maintenance related reasons. The yearly availability is higher than in many of the previous eight years (85.11%, 36.17%, 34.57%, 89.69%, 88.15%, 75.68%, 66.67%, 58.65%) at 74.10% for this reporting year with most of the forced unavailability due to maintenance to troubleshoot and implement a new means to check the PC level trip, to repair the secondary cooling flow meter and restore the trip on well water loss of flow, to repair the period trip bistable card, to repair the secondary heat exchanger sample line and to address the trip on high southeast fuel box temperature indication.

Overall the availability represents about average availability recorded for the past ten or more reporting years. This is due to having no large forced outages until near the end of the year. Of the 49½ days forced outage time, maintenance to implement a new means to check the PC level trip (4 days in March 2005), to troubleshoot and then repair the secondary cooling flow meter and well water flow trip (8 days in July 2005), to repair the period trip bistable card (16⅞ days in July/August 2005), to repair the break in the secondary heat exchanger sample line (3⅜ days in August 2005) and to address trips on the high Primary Coolant southeast fuel box outlet temperature (18⅛ days in June-August 2005) involved significant forced outages. No other forced outage involved even one full day. There was one significant planned outage this year to perform the annual calibration and calorimetric heat balance (A-2 Surveillance) in March 2005 (6⅞ days). Other than these outages, the remainder of the year saw the usual variety of maintenance activities and equipment failures though the 7½ days administrative shutdown in September 2004 for two hurricane watches was unusual. It is hoped that quality maintenance will assure high availability in the 2005–6 reporting year.

TABLE III-4

UFTR AVAILABILITY SUMMARY
(September 2004 – August 2005)

Month	Availability	Days Unavailable	Primary Cause of Lost Availability
			(F) Forced (P) Planned
September 2004	69.17%	9.250 days	Maintenance (P) to replace the diesel generator contactor and timing coil to restore the backup power availability to the reactor (1⅝ days). Maintenance (P) to replace the fan belts on the stack dilute fan and grease the bearings (⅛ day). Administrative shutdown for potential effects of Hurricane Frances and Hurricane Jeanne (7½ day).
October 2004	95.97%	1.250 days	Maintenance (P) to check out tapping noises (⅛ day). Maintenance (P) to replace the gear box on the east area radiation monitor chart recorder to correct periodic slippage (⅛ day). Administrative shutdown to addressing chill water line replacement activities in the west lot (1 day).
November 2004	92.08 %	2.375 days	Maintenance (P) to replace the filter/demineralizer cartridges on the shield tank water purification system (¼ day). Maintenance (P) to refill the primary coolant storage tank (⅛ day). Administrative shutdown for the Thanksgiving holiday (2 days).

TABLE III-4

UFTR AVAILABILITY SUMMARY
(September 2004 – August 2005)

Month	Availability	Days Unavailable	Primary Cause of Lost Availability
December 2004	85.89%	4.375 days	Maintenance (F) to clean sticking internals on the secondary flow meter (⅛ day). Maintenance (P) to lubricate a non-responsive city water flow meter valve actuator (⅛ day). Maintenance (P) to replace a worn gear to correct slippage on the AMS[4] chart recorder drive (⅛ day). Administrative shutdown for the Christmas and New Year's holidays (4 days).
January 2005	92.74%	2.250 days	Maintenance (F) to fix a paper jam on the two-pen recorder (⅛ day). Maintenance (P) to tighten belts and check motor on dilute fan motor relative to slightly reduced rpm indication (⅛ day). Administrative shutdown for the New Year's and Martin Luther King Jr. holidays (2 days).
February 2005	99.55%	0.125 days	Maintenance (P) to refill the primary coolant storage tank (⅛ day).
March 2005	64.11%	11.125 days	Maintenance (F) to replace the deep well pump 60 amp fuses and verify effectiveness (⅜ day). Maintenance (F) to design, install, check and implement a new means to check the trip on loss primary coolant level (4 days). Maintenance (F) to repair unresponsive red pen on the two-pen recorder (¼ days).

TABLE III-4

UFTR AVAILABILITY SUMMARY
(September 2004 – August 2005)

Month	Availability	Days Unavailable	Primary Cause of Lost Availability
March 2005 *(continued)*			Maintenance (P) to make various voltage checks and minor adjustments as part of the annual UFTR Nuclear Instrumentation Calibration Check and Calorimetric Heat balance (A-2 Surveillance) including correcting switched temperature points for thermocouples #6 and #7 (6½ days and concurrent for ⅜ day).
April 2005	99.17%	0.250 days	Maintenance (P) to refill the primary coolant storage tank (⅛ day).
			Maintenance (P) to replace certain batteries in the security system following low voltage indication (⅛ day).
May 2005	93.15%	2.125 days	Maintenance (F) to clean and assure proper operation of a sticking secondary flow meter (⅛ day).
			Maintenance (P) to replace failed flow meter in the city water cooling line (1½ days).
			Administrative shutdown for the Memorial Day holiday (½ day).
June 2005	79.17%	6.250 days	Maintenance (F) to address the trip on high temperature for the SE Fuel Box exit temperature (1⅛ days).
			Maintenance (P) to refill the primary coolant storage tank (¼ day).
			Administrative shutdown for the Facility Director's absence (4⅞ days).

TABLE III-4

UFTR AVAILABILITY SUMMARY
(September 2004 – August 2005)

Month	Availability	Days Unavailable	Primary Cause of Lost Availability
July 2005	12.50%	27.125 days	Maintenance (F) to clean and repair the secondary flow meter and restore the trip on well water flow (8 days). Maintenance (F) to repair the period trip bistable card (3 days). Maintenance (F) to address the trip on high temperature for the SE Fuel Box exit temperature (3⅛ days). Administrative shutdown for the Facility Director's absence (13 days and concurrent for 1¼ days).
August 2005	5.65%	29.250 days	Maintenance (F) to address trips on high temperature (12 days). Maintenance (F) to repair the secondary heat exchanger sample line (3⅜ days). Maintenance (F) to repair the period trip bistable card (13⅞ days).

TOTAL ANNUAL UNAVAILABILITY (Availability at 73.767%):	95.750 days	=	26.233%
1. TOTAL FORCED UNAVAILABILITY:	49.500 days	=	13.562%
2. TOTAL PLANNED UNAVAILABILITY:	11.375 days	=	3.116%
3. TOTAL ADMINISTRATIVE UNAVAILABILITY:	34.875 days	=	9.555%

NOTE 1. This availability summary neglects all minor unavailability for periods smaller than one-eighth day. In most cases these periods are for much less than an hour as some minor problem is corrected, such as replacing chart paper on an area radiation detector or a light bulb in an indicator, usually during or after a preoperational checkout. This availability summary also neglects unavailability for scheduled tests and surveillances except where noted when maintenance becomes necessary.

TABLE III-4

UFTR AVAILABILITY SUMMARY
(September 2004 – August 2005)

Month	Availability	Days Unavailable	Primary Cause of Lost Availability

NOTE 2. The 95.750 days total unavailability in the 2004–5 reporting year is one of the lowest in recent years with the forced outage rate at 49.500 days versus 26.375 days, 217.50 days and 235.00 days in the previous three reporting years and with the planned outage rate at only 11.375 days versus 13.000 days, 13.625 days and 1.250 days in the previous three reporting years. There were five forced outages to exceed three days for designing, installing, checking and implementing a substitute method of checking the primary coolant level trip (4 days), for repairing the secondary flow meter and restoring the trip on well water flow (8 days), to address the failure of the period trip bistable card (16.875 days), to repair the secondary heat exchanger sample line break (3.375 days) and to address the two trips on high southeast fuel box outlet temperature indication (18.125 days). The total unavailability time is for maintenance for repairs, delays awaiting parts arrival, trip evaluations, etc. plus 34.875 additional days of administrative shutdown compared with 15.5 days, 0.5 days and 4.00 days in the previous three reporting years delineated in this table for holidays, potential external events, and associated personnel vacations or unavailability of management to approve operating where the reactor was or could have been made operational if needed. With no full-time technical staff members for the year and only the Director licensed from late March 2005 to late August 2005, the last category for administrative shutdown is larger than in most years and includes 7.5 days for two hurricane watches.

NOTE 3. It should be noted that only category 1 and 2 unavailability values were listed under repair and maintenance related (loss of reactor) unavailability prior to the 1991–92 year. The total unavailability in these categories has tended to go in cycles partially dependent on effectiveness of previous maintenance plus the wear out of equipment for which there is no on-hand spare. This was true of the outages for the failed fission chamber and the failed deep well pump in the 2002–3 reporting year and somewhat for the sticking control blade problem in the 2003–4 reporting year. It is also true for the period trip bistable card failure and the secondary flow meter repair in this 2004–5 reporting year. The lost availability for administrative reasons has shown some variation in earlier reporting years from as many as 23.50 days to as low as 0.5 days but is quite high at 34.875 days in the 2004–5 reporting year.

TABLE III-5A

UNSCHEDULED TRIPS
(September 2004 – August 2005)

After three unscheduled trips occurred in the first three months of the 1989–90 reporting year, none occurred during the 1990–91 reporting year; in the 1991–92 reporting year, three unscheduled trips occurred in November 1991, December 1991 and May 1992. It is worth noting that in the 1992–93 reporting year, the first unscheduled trip occurred in March 1993 and was the first experienced in nearly ten months, the second unscheduled trip occurred in August 1993. As with two of the three trips in the 1991–92 reporting year, one of these trips was due to an electrical transient while the other was due to inadvertent operator action, as was the third trip in the 1991–92 reporting year, with neither considered to have significantly affected reactor safety or the health and safety of UFTR personnel or the public. All safety systems responded properly for each trip and a full review was conducted prior to restart in each case with the second trip considered to be promptly reportable. After having no unscheduled trips during the 1993–94 reporting year, the UFTR experienced two unscheduled trips during the 1994–95 reporting year as it did again in the 1995–96 reporting year. The UFTR experienced no unscheduled trips during the 1996–97 reporting year. It is also worth noting that the two trips described and evaluated in this table in the 1995–96 reporting year were the only unscheduled trips for over three reporting years until July 30, 1999 and only the second trip was evaluated to be due to equipment failure due to faults in the Safety Channel 2 loss of high voltage sensing circuit. For the 1998–99 reporting year, there was only one trip evaluated as due primarily to a somewhat more restrictive loss of voltage setting on the power supply for Safety Channel 2 plus a much-taxed electrical distribution system due to a heat wave. This single unscheduled trip was described and evaluated in the single entry in this table for the 1998–99 reporting year.

Again for the 1999–2000 reporting year, there was only one unscheduled trip evaluated as due to a campus-wide power outage for less than about one minute which resulted in a full trip which was not caused by any facility-related equipment or equipment malfunction with all protection and safety systems responding properly. This single unscheduled trip was described and evaluated in the single entry in this table for the 1999–2000 reporting year report.

Although a number of failed components were replaced to complement replacement of degraded components along with preventive cleaning and repair of circuit connections in the 1989–90 reporting year, as well as in the past eleven years, these efforts clearly have represented time well spent with very few trips due to facility equipment failure in the last nine years and none during the past 1996–97 and 1997–98 reporting years until July 30, 1999. The trip in the 1999–2000 reporting year on February 9, 2000 was again not due to facility equipment malfunction.

For the 2000–2001 reporting year, there were only three unscheduled trips; all are addressed in the 2000–2001 table. The first on September 12, 2000 was a full trip at full power due to an area power outage, again not due to facility equipment malfunction. The second trip (also a full trip) on July 20, 2001 was due to the operator inadvertently pushing the power off versus the automatic to manual control button in preparation for commencing shutdown from full power, again not due to facility equipment malfunction. Finally, the third full trip, also at full power, was due to a failure in the detector systems part of the wider range drawer and was due to facility equipment malfunction, troubleshooting for which was continuing at year's end per entry 3 in the Table III -5A for the 2000 – 2001 year.

TABLE III-5A

UNSCHEDULED TRIPS
(September 2004 – August 2005)

For the 2001–2 reporting year, there was only one unscheduled trip plus one carried over from the previous year; both are addressed in the 2001–2 table. The first carried over from July 26, 2001 of the previous year was a full trip at full power due to a failure in the detector systems part of the wider range drawer and was due to facility equipment malfunction. The second trip (a blade drop, process trip) on February 22, 2002 was due to a power surge interrupting power to the temperature/monitor/recorder resulting in a process trip on high temperature; it was not due to equipment failure.

For the 2002–3 reporting year, there was only one unscheduled trip. This full trip occurred during startup on August 4, 2003 due to noise generated from the Regulating Blade bottom limit switches as updrive of the Regulating Blade was begun. A modification to suppress noise generation prevented recurrence of this trip as noted in this table in the 2002–3 report as this full trip was somewhat attributable to faulty equipment.

For the 2003–4 reporting year, there was only one unscheduled trip as addressed in the 2003–4 annual report. This blade drop, process trip occurred during power reduction for temperature coefficient measurements on April 6, 2004 due to hysteresis effects in the trip on loss of secondary cooling which was evaluated as acceptable in the only table entry for the 2003–4 reporting year.

For the 2004–5 reporting year, there were three unscheduled trips. The first trip was a blade drop, process trip due to loss of secondary coolant at full power due to fuse failure on the deep well pump power with all safety systems responding properly as noted in entry 1 from this table. The second and third trips were essentially identical events resulting from a failure in the temperature monitoring channel (thermocouple or connective wiring and connections) for fuel box outlet point #3 (Southeast Fuel Box Outlet) as discussed in entries #2 and #3 for this table.

TABLE III-5A

UNSCHEDULED TRIPS
(September 2004 – August 2005)

Number	Date	Description of Occurrence
1.	7 Mar 05	On March 7, 2005, reactor startup was begun at 0911 hours to support the annual UFTR Nuclear Instrumentation Calibration Check and Calorimetric Heat Balance (A-2 Surveillance) with 100 kW full power reached at 0931 hours. Subsequently, a blade drop trip occurred at 1107 hours while the reactor was critical at 100 kW and running for approximately 1.5 hours in automatic control. The SRO operator noted the "SEC PRESS" indicator lit on the scram annunciator after the "Well Warning" and "Flow Scram" indicators energized, indicating loss of or low secondary water flow. In accordance with Technical Specifications Section 2.2, approximately 10 seconds later the "SEC PRESS" trip came in and tripped the reactor. At the same time the bottom indicator of the secondary well pump on/off switch energized, indicating a loss of power at that point. MLP #05-04 was then opened and the well pump fuses were immediately checked for continuity. The southern most 60 amp well pump fuse was found to be blown and hot to the touch. The other fuses were working properly (continuity check – 0.2 ohms). All three fuses were replaced and documented in the semiannual replacement of deep well secondary pump fuses (S-9 Surveillance). The overload reset button was also reset. The secondary well pump was then energized and left running for approximately 1.5 hours. No abnormalities were observed during this time and the well pump was secured. UFTR Form 0.6A (Unscheduled Reactor Trip Review and Evaluation) was also completed to approve restart. The subsequent restart and operation for over 7 hours on March 8 to complete the A-2 Surveillance was also successful. Based on the indications at the time of the blade drop trip, the well pump 60 amp fuse failed. The fuse failure resulted in a reduction in water flow and an increase in amperage in the other two phases of electrical power. The reduction in water flow caused the "SEC PRESS" trip after a 10-second delay. The increase in amperage of the other two phases caused the overload circuit to trip and remove power to the secondary well pump. All safety and control systems were noted to have operated correctly and in accordance with the Tech Specs for the trip from a known cause. Based on this fact, this blade drop trip is evaluated as not promptly

TABLE III-5A

UNSCHEDULED TRIPS
(September 2004 – August 2005)

Number	Date	Description of Occurrence
		reportable as defined in Technical Specifications Section 6.6.2. This event is also evaluated as having negligible impact on reactor safety or on the health and safety of the public or reactor facility staff. The completed UFTR Form 0.6A (Unscheduled Reactor Trip Review and Evaluation) is Attachment I to the March 2005 monthly report. A memorandum describing the reactor trip event is Attachment II to the March 2005 monthly report.
2.	27 Jun 05	On June 27, 2005, the reactor was started up at 1553 hours and run at full power starting at 1611 hours using the alternate city water cooling mode to complete the quarterly Radiological Survey of Restricted Areas (Q-5 Surveillance). The Q-5 Surveillance was completed at 1645 hours and the reactor remained at full power to irradiate samples for UCF researchers. Subsequently, a high primary coolant temperature spike at 921.9° F on the SE Fuel Box (Point #1) was received resulting in a blade drop trip at 1653 hours. With all control and safety systems noted to be responding properly, the reactor was secured at 1653 hours. The temperature of 921.9° F on the SE Fuel Box (Point #1) was noted to return to normal again by spiking down at about 1659 hours. Because of the essentially instantaneous >800° F change in temperature on the SE Fuel Box with all other indications normal, this event was evaluated to be due to a failure in the temperature monitoring system rather than to any actual condition requiring a high temperature trip. In particular, the core bulk outlet temperature at Point #8 was noted to be unchanged during the event. Subsequently, the occurrence was evaluated to be due to an intermittent fault in the temperature monitoring system, possibly exacerbated by the ~12° F higher primary coolant exit temperature experienced at full power in the city water cooling mode. Under MLP #05-13, opened to eliminate the computer or thermocouple as the source of the problem, a higher temperature was simulated. Basically, the use of a simulated high temperature (147° F) with no trip proved that the computer/virtual display device works properly. Next, to identify the thermocouple as failed or failing, resistance readings were made on the thermocouple lines for the SE Fuel Box as well as for all other nine (9) points monitored. The results show that, if the SE Fuel Box thermocouple did separate, or any connection to the thermocouple separated, then it has restored itself mechanically. This result indicates the fault is intermittent and difficult to isolate. At this point, UFTR

TABLE III-5A

UNSCHEDULED TRIPS
(September 2004 – August 2005)

Number	Date	Description of Occurrence
		Form SOP-0.6A (Unscheduled Reactor Trip and Evaluation) was completed, MLP #05-13 was closed out, and the reactor approved for restart subject to a successful daily checkout. Since this occurrence was from a known cause, this trip is not considered promptly reportable, though an NRC Project Manager was told of the occurrence on June 29, 2005 in relation to other issues. In addition, it is evaluated as having had negligible effect on reactor safety and no effect on the health and safety of the public with a successful daily preoperational check completed on June 30, 2005 with no further problems noted at month's end. Completed Form SOP-0.6A (Unscheduled Reactor Trip and Evaluation) is Attachment I to the June 2005 monthly report.
3.	22 Jul 05	Following the high temperature trip (spurious) caused by indicated high temperature on the southeast fuel box outlet thermocouple indication which occurred after operating for 42 minutes at 100 kW (72.333kWh) on June 27, 2005, the event recurred after operating at 100 kW for 27 minutes (47.333 kWh) on July 22, 2005. On July 22, 2005, the reactor was started up at 1555 hours and run at full power starting at 1616 hours using the alternate city water cooling mode to complete an update of part of the quarterly Radiological Survey of Restricted Areas (Q-5 Surveillance) following rearrangement of the rabbit system shielding to reduce radiation levels. The reactor was simultaneously being run to conclude irradiation of samples for UCF researchers. The partial Q-5 Surveillance was essentially completed at 1643 hours when a high primary coolant temperature spike at 921.9° F on the SE Fuel Box (Point #1) was received resulting in a blade drop trip at 1643 hours. With all control and safety systems noted to be responding properly, the reactor was secured at 1644 hours. The temperature of 921.9° F on the SE Fuel Box (Point #1) was noted to return to normal again by spiking down at about 1649 hours. Because of the essentially instantaneous >800° F change in temperature on the SE Fuel Box with all other indications normal, this event was again evaluated to be due to a failure in the temperature monitoring system rather than to any actual condition requiring a high temperature trip, especially since the core bulk outlet temperature at Point #8 was noted to be unchanged during the event. Subsequently, the occurrence was again evaluated to be due to an intermittent fault in the temperature monitoring system, possibly

TABLE III-5A

UNSCHEDULED TRIPS
(September 2004 – August 2005)

Number	Date	Description of Occurrence
		exacerbated by the ~12° F higher primary coolant exit temperature experienced at full power in the city water cooling mode. The use of a simulated high temperature (147° F) with no trip proved that the computer/virtual display device works properly following the earlier trip. Following the earlier trip, resistance readings were made on the thermocouple lines for the SE Fuel Box as well as for all other nine (9) points monitored. The results showed that, if the SE Fuel Box thermocouple did separate, or any connection to the thermocouple separated, then it has restored itself mechanically. These results had been taken to indicate that the fault is intermittent and difficult to isolate. Nevertheless, it was noted that the system had returned to normal. Since this occurrence was from a known cause, this trip is not considered promptly reportable, though NRC Project Manager Al Adams was told of the occurrence on July 22, 2005 because an instant SRO examination had been scheduled for July 25–26, 2005. In addition, it was evaluated as having had negligible effect on reactor safety and no effect on the health and safety of the public as with the previous occurrence. Therefore, it was decided to consider limited UFTR operations before undertaking further maintenance efforts. On July 25, 2005, the RSRS Executive Committee met to review UFTR status and consider possible future operations at limited power levels following the recurrence of the high temperature trip on July 22 after first occurring on June 27, 2005 while operating on lower flow city water secondary cooling resulting in somewhat higher core fuel box outlet water temperatures of 110°–115° F due to unavailability/ unreliability of the secondary well water cooling mode. It was noted that these temperatures are well within the allowable operating temperatures for the cooling water. In both cases the SE fuel box was the indicated failure, though not the hottest box, with the temperature instantaneously spiking to indicate 921.9° F and returning to normal in 5–6 minutes. Because the system had returned to normal and remained so, the RSRS Executive Committee was convened where it was requested that several lower power operations be approved (providing all indications remain satisfactory) with power levels limited to 25 kW to preclude reaching the temperature where the trip might be initiated.

TABLE III-5A

UNSCHEDULED TRIPS
(September 2004 – August 2005)

Number	Date	Description of Occurrence
		The Executive Committee evaluated various input information including the possibility that the high temperature trip is the result of a failing thermocouple or electrical connection and clearly is not an actual temperature transient; that all safety systems have responded properly for both trips; that the likelihood is the intermittent failure will recur at increasing frequency; that the intermittent failure may be due to thermocouple/electrical connections in core and is exacerbated by increased temperature; that the approval for further reduced power operations would be limited in time; that there are multiple indicators to provide reliable reactor core status even after a thermocouple/ temperature monitoring channel failure; that reactor operations are needed for an SRO-trainee's NRC license examination; and finally, that there may be a need for two operators to facilitate completion of surveillances after maintenance to repair the temperature monitoring channel should a core entry be needed. After extensive discussions, the Executive Committee voted to approve operations to 25 kW but for only one usage for the SRO-trainee's licensing examination (understood to include prior successful performance of the requisite weekly and daily preoperational checkouts) with no further critical/startup operations allowed until further repairs would be implemented. One committee member also recommended eliminating all possible sources of the trip problem external to the core before undertaking the work to unstack the core. The Facility Director was also to discuss this approval with the NRC Project Manager and the NRC Inspector to assure that they were not opposed to this limited operation. This was accomplished effectively in two conversations with Al Adams who also contacted the NRC License Examiner to assure the license examination for the SRO-trainee was rescheduled for August 1–2, 2005. Following successful completion of the weekly preoperational checkout on July 25, 2005, and the RSRS Executive Committee agreement to a single low power run, UFTR Form SOP-0.6A (Unscheduled Reactor Trip Review and Evaluation) was completed and the reactor approved for the low power operation subject to a successful daily preoperational check which was delayed until July 28, 2005 due to an unrelated period trip bistable card failure (see Section II.C.2 of the July 2005 monthly

TABLE III-5A

UNSCHEDULED TRIPS
(September 2004 – August 2005)

Number	Date	Description of Occurrence
		report). The completed UFTR Form SOP-0.6A is Attachment II to the July 2005 monthly report. The Minutes of the RSRS Executive Committee meeting at which the limited restart was approved are Attachment III to the July 2005 monthly report. At the end of July, the reactor had successful weekly and daily preoperational checks completed and was prepared for the low power operation for the license examination which was completed on August 2, 2005. Subsequently, under Maintenance Log Page #05-16, opened after the second trip, and using the half-splitting technique, the panel which contains wiring for the thermocouple system was identified in the equipment pit. This panel additionally contains main power lines (but no distribution blocks) for all pit equipment. This arrangement was considered a possible cause of electromagnetic interference and a possible source of the problem with the high temperature indication on the southeast fuel box outlet thermocouple, but it was considered unlikely as the system has operated for a number of years without a problem of the type seen with this high temperature indication. The six in-core thermocouples are all connected to a terminal block within the panel in the pit. These are the only connections on the block. This block has 2 connectors per side, per thermocouple, with 2 sides and therefore contains 12 total connections. Half of these are constantan and the other half copper. This is consistent with Type T (copper-constantan) thermocouple wiring considerations. Under MLP #05-16, mechanical agitation of the system was begun on August 2, 2005 with light initial agitation defined as slowly moving the individual cables as lightly as possible without interfering with the other cables. The proximity of the incoming (from console) and outgoing (to core) lines prevented perfect isolation of the individual lines. Moving one line could (and in some cases did) contact the other lines regardless of the care with which the chore was performed. The result of the agitation was to generate identical and/or similar temperature spikes on several of the temperature monitoring points including the southeast fuel box outlet (Point #3) and even to a lesser extent on the secondary outlet temperature (Point #10).

TABLE III-5A

UNSCHEDULED TRIPS
(September 2004 – August 2005)

Number	Date	Description of Occurrence
		These results were obtained over the course of several hours, and in some cases were very repeatable. All signal anomalies occurred more than one time. Some anomalies were easily repeated with the same wire motion, others were not. These anomalies were representative of the initial fault which was seen on the southeast fuel box outlet including a spike up and resultant but delayed return to normal. The secondary outlet could not be duplicated. Commonalities considered include the type and age of wire, connection types, connection locations, and path to that location. A general concern at this time was a faulty connection at the block or failure in the wire (hairline fracture) possibly from manufacturing. Subsequently, on August 2, 2005, all connections were reseated (detached and reattached) on the connection block. On August 3, 2005, using a heat gun, the junction box and surrounding wires were heated for 30 minutes. No obvious change occurred during that period, but concern for the insulation necessitated removal of the heat source. A more gradual heat source (1500 W max space heater) was identified and this was used to heat and maintain heat to the area for approximately 2 hours at nearly 90 °F. This is not as hot as the system would probably get during full power operations, especially using city water secondary cooling as was being used at the time of both trips on June 27 and July 22, 2005. During this period no obvious signal changes occurred. Only with mechanical agitation could any changes be induced and these were much reduced, apparently due to the reseating on the connection block completed on August 2, 2005. A substandard mechanism of attachment for several of the spade terminal lugs-to-wire connections was found. The wires were attached by mechanical means employing only a hook through a hole which does not provide a permanent electrical connection. It is possible that the wire, with little heating, could be moving out of contact with the spade lug. It was recommended that these junctions be soldered and the system retested. Zinc chloride flux or equivalent was required to solder the constantan joints. At this point the RSRS Executive Committee was convened again, on August 9, 2005. They were requested, upon completion of the

TABLE III-5A

UNSCHEDULED TRIPS
(September 2004 – August 2005)

Number	Date	Description of Occurrence
		soldering and successful retesting, to approve reactor operation at full power with either form of secondary water cooling to verify successful maintenance. After extensive discussions, the Executive Committee voted to approve the 100 kW test run. The resoldering operation was finally completed on August 17, 2005. In addition, thermocouple wire was located and ordered but no replacement thermocouples were located. After August 18, 2005, maintenance time was spent addressing the period bistable failure (see Section II.C.2). The completed UFTR Form SOP-0.6A for the July 22, 2005 trip is Attachment I to the July 2005 monthly report. The Minutes of the August 9, 2005 RSRS Executive Committee meeting at which the limited restart following determination of temperature monitoring wires was approved constitute Attachment I to the August 2005 monthly report which includes the memorandum which outlined the proposed maintenance and subsequent reactor operation to verify operability. At reporting year's end (end of August, 2005), the reactor was nearly ready for the power run to check repair of the thermocouple monitoring point connections.

TABLE III-5B

SCHEDULED TRIPS
(September 2004 - August 2005)

There were no scheduled trips performed for experimental or training purposes during the last three reporting years and only one scheduled trip performed for experimental purposes during the 1998–99 reporting year. That trip was the first scheduled trip in a number of years. Part of the reason for this general lack of scheduled trips is the failure to schedule any large utility operator training programs where such trips are a designed part of the training program. It was anticipated that some training trips would be included in the ENU-5176L Reactor Operations Laboratory course offered during the 1996–97 or 1997–98 reporting years to demonstrate similarities and differences in power response for trips versus normal shutdown as well as in various student laboratory exercises to demonstrate rapid decay and recovery of stack count rate with power reduction and increase as part of Argon-41 stack effluent measurement exercises, but this did not occur. The nearly yearlong outage for the 1998–99 reporting year again precluded such training trips. It was expected these training trips might occur in the 1999–2000 reporting year, the 2000–2001 reporting year, 2001–2, 2002–3 or 2003–4 reporting year but they did not. During the 2004–5 reporting year the NRC license examiner requested a trip at 9 kW as part of the operations portion of the RO examination on August 2, 2005. After proper review prior to the trip, a manual scram was performed with all systems responding properly. It is expected that one or more scheduled trips might occur in the 2005–6 reporting year, especially to determine some of the HEU response parameters relative to the HEU to LEU fuel conversion. Such trips can also be used to provide training in control room presence and awareness of changing conditions and responses in training UFTR operator license candidates and may be utilized as time permits in the next reporting year. Since there was one scheduled trip during this reporting year, there is a single entry in this table.

Number	Date	Description of Occurrence
1.	2 Aug 05	During the operating portion of the NRC administered licensing exam, while at 9 kW steady state power level, the examiner requested that the operator license candidate trip the reactor. After checking with the operator of record, the trainee performed a manual scram with all systems responding properly. UFTR Form SOP-0.6A (Unscheduled Reactor Trip Review and Evaluation) was completed for this scheduled trip to assure proper evaluation and documentation and is Attachment VII to the August 2005 monthly report availability at the facility.

TABLE III-6

LOG OF UNUSUAL OCCURRENCES
(September 2004 – August 2005)

During this reporting year there were no events considered to have compromised reactor safety or the health and safety of the public. Eight events classified as unusual occurrences, none as promptly reportable potential abnormal occurrences are listed in this table. These events are described below as they deviated from the normal functioning of the facility and are included here as the most important such deviations for the reporting year. Unscheduled shutdowns are covered here as well, as one occurred here this year (occurrence #3). Unscheduled trips are also addressed here though they are detailed in Table III-5A along with corrective and preventive maintenance and surveillances implemented in response to the trips where applicable; three such trips occurred during this reporting year for a process trip for loss of secondary cooling (occurrence #4) and two for process trips on indicated apparent high temperature on the southeast fuel box outlet monitoring channel (Point #3) (occurrences #6 and #7).

Of the eight occurrences this year two did not involve some equipment failure, inadequacy or other event. Occurrences #1 and #2 were both for hurricane watches for Hurricane Frances beginning September 3, 2004 and for Hurricane Jeanne starting on September 24, 2004. Both hurricanes were centered sufficiently far enough away and sufficiently weakened that they had minimal impact on the facility site as tropical storms though the emergency flood procedure (SOP-B.4) was implemented to address the potential to impact the facility in both cases. All remaining six occurrences involved some equipment failure, inadequacy or other event. The most significant occurrences were the three process blade drop reactor trips described in occurrence #4 which involved a process trip for loss of secondary cooling due to secondary well pump failure while occurrences #6 and #7 were for process trips on apparent high temperature on the southeast fuel box outlet monitoring channel (Point #3) due to thermocouple and/or connective wiring failure. The next most significant event would be the unscheduled shutdown due to inadvertent switching of temperature monitoring points (occurrence #3). Occurrence #5 to address failure of a test mechanism used to test the trip on loss of primary coolant level was a minor item; similarly, occurrence #8 to address repairing of the failed period trip bistable card was discovered during a preoperational check and so has little safety significance though it accounted for several weeks of forced outage time to obtain, verify and implement a replacement card.

Overall, none of these eight occurrences is considered to have had significant impact on the safety of the reactor or on the health and safety of the public. In addition, all have been reviewed to assure adequate consideration of their effects with none officially reported promptly to the NRC, though all were reported for information purposes at some point. All were also reported in periodic updates to the NRC.

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(September 2004 - August 2005)

Number	Date	Description of Occurrence
1.	3 Sep 04	On Friday, September 3, 2004, as Hurricane Frances was predicted to have near certain effects on the Gainesville area, the weather predictions were for up to 10–12 inches of rain. The University of Florida was shut down at noon on Friday and for the weekend in expectation of possible tropical storm effects. It was decided to implement the UFTR flooding procedure and, although not required, insert one hurricane rod into the center vertical port per SOP-B.4 and to secure much of the facility's electronic equipment including computers and HPGe detectors as well as to secure sources of water entry at the west lot doors and move equipment above floor level. The hurricane rod insertion was accomplished at 1550 hours on September 3, 2004 by W.G. Vernetson. Subsequently, Hurricane Frances had significant tropical storm effects on the Gainesville area with many downed trees, much flooding and much loss of electrical power. On Sunday, September 5, the facility was checked in response to a security alarm in the afternoon, and then also checked for storm damage with none found at that time, so the facility was returned to security with no further problems noted. Subsequently, the Director together with RSRS Acting Chairman W.S. Properzio, in response to a UPD report, noted a damaged tree was leaning on the southwest corner of the Nuclear Science Building (NSB) but with no effect on the Reactor Building. Water was also noted to be entering the doorway of the NSB southwest entrance due to a drain plugged with storm debris which was removed by emergency personnel on numerous occasions throughout the weekend. On Monday, September 6, the university was closed for the Labor Day holiday and remained closed on Tuesday, September 7, primarily due to the number of students, faculty and staff without home electrical power as the campus was little affected by power outages. The facility was visited by the Director on September 7 and, though not removed from security, it was noted to be unaffected by the hurricane although the southwest entrance of the NSB was still showing water inside from the clogged drain. In response to an earlier call on September 3, an NRC inspector was informed via (voice mail) message that the facility was in good shape. Subsequently, on September 7, 2004, at 1215 hours, the flooding condition was terminated. On Wednesday, September 8, the university was reopened whereupon the hurricane rod was then removed by SRO W.G. Vernetson and RCO D.L. Munroe at 1010 hours and returned to its storage location after

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(September 2004 - August 2005)

Number	Date	Description of Occurrence
		assuring it was not contaminated. The facility was then returned to normal operations with no impact on reactor safety and no impact on the health and safety of the public
2.	24 Sep 04	On Friday, September 24, 2004, Hurricane Jeanne was predicted to have near certain effects on the Gainesville area with up to 10 inches of rain possible. In response, the University of Florida was shut down for the weekend in expectation of impending tropical storm effects. It was decided to implement the flooding procedure and, although not required, insert one hurricane rod in the center vertical port per SOP-B.4 and to secure much facility electronic equipment including computers and HPGe detectors and secure sources of water entry at the west lot doors and move equipment above floor level. This rod insertion was accomplished at 1640 hours on September 24 with training for RO-trainee M. Berglund. Subsequently, Hurricane Jeanne began to affect Gainesville overnight with near tropical storm force winds. The facility was checked in response to a security alarm in the afternoon on Saturday, September 25 with no effects noted as the facility was returned to security with no problems noted. Although tropical storm force winds were essentially over by midday, Sunday, September 26, the university remained closed on Monday, September 27, primarily due to the number of students, faculty and staff without electrical power and with many sheltered on campus which was little affected by electrical power outages. The facility was visited by the Director on September 27 and, though not removed from security, it was noted to be non-affected by the hurricane's tropical storm force winds and rain. Subsequently, on September 27, 2004 the flooding condition was terminated at 1640 hours. The university was reopened on Tuesday, September 28, whereupon the hurricane rod was removed by SROs W.G. Vernetson and B. Shea at 1215 hours and then prepared for return to its storage location after assuring it was not contaminated. The facility returned to normal operations with no impact on reactor safety and no impact on the health and safety of the public
3.	3 Mar 05	On March 3, 2005, following performance of various initial adjustments and checks as part of the annual UFTR Nuclear Instrumentation Calibration Check and Calorimetric Heat Balance (A-2 Surveillance), a reactor startup was commenced at 1201 for the

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(September 2004 - August 2005)

Number	Date	Description of Occurrence
		first power run to allow generation of sufficient neutron level per SOP-E.4, Step 7.1.9. At 1223 hours, with reactor power at 30 kW, the operator noted primary coolant inlet temperature reading 77.4 °F and primary outlet temperature reading 74.2 °F indicating the two points had apparently been reversed during the prior adjustments and checks. An unscheduled shutdown was commenced at 1223 hours with the reactor shut down and secured at 1225 hours and all systems responding properly. Subsequently, under MLP #05-03 applicable for the A-2 Surveillance adjustments and checks, the thermocouple connections were checked; Point #6 (NE box) and Point #7 (inlet) thermocouple connections were found to have been switched. The connections were then placed in the correct positions and all connections verified (Points #1–#10) by the SRO and independently by an SRO-trainee. Following a successful daily checkout, UFTR Form SOP-0.6B (Unscheduled Reactor Shutdown Review and Evaluation) was completed and restart to continue the A-2 Surveillance (Step 7.1.9) was approved. It was noted that the switched thermocouple connections were realized quickly by an alert operator as soon as possible based on temperature differences; the misconnected switches had no impact on safety since all temperature points provide the same trip impact so there was negligible impact on safety due to the error. Subsequently, a second daily checkout was completed on March 4, 2005. During the subsequent restart, the backup Honeywell temperature monitor was noted to be producing erroneous readings but it was reinitialized to restore proper indication. The 6-hour run at full power was successful with shutdown commenced at 1527 hours on March 4. The reactor was shut down and secured with no problems noted at 1529 hours. During the unscheduled shutdown, all safety and control systems were noted to have operated correctly and in accordance with the Tech Specs. Based on this fact, this unscheduled shutdown is evaluated as not promptly reportable as defined in Technical Specifications Section 6.6.2. This event is also evaluated as having negligible impact on reactor safety or on the health and safety of the public or reactor facility

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LOG OF UNUSUAL OCCURRENCES
(September 2004 - August 2005)

Number	Date	Description of Occurrence
		staff. The completed UFTR Form 0.6B (Unscheduled Reactor Shutdown Review and Evaluation) is Attachment III to the March 2005 monthly report.
4.	7 Mar 05	On March 7, 2005, reactor startup was begun at 0911 hours to support the annual UFTR Nuclear Instrumentation Calibration Check and Calorimetric Heat Balance (A-2 Surveillance) with 100 kW full power reached at 0931 hours. Subsequently, a blade drop trip occurred at 1107 hours while the reactor was critical at 100 kW and running for approximately 1.5 hours in automatic control. The SRO operator noted the "SEC PRESS" indicator lit on the scram annunciator after the "Well Warning" and "Flow Scram" indicators energized, indicating loss of or low secondary water flow. In accordance with Technical Specifications Section 2.2, approximately 10 seconds later the "SEC PRESS" trip came in and tripped the reactor. At the same time the bottom indicator of the secondary well pump on/off switch energized, indicating a loss of power at that point. MLP #05-04 was opened and the well pump fuses were immediately checked for continuity. The southern most 60 amp well pump fuse was found to be blown and hot to the touch. The other fuses were working properly (continuity check – 0.2 ohms). All three fuses were replaced and documented in the semi-annual replacement of deep well secondary pump fuses (S-9 Surveillance). The overload reset button was also reset. The secondary well pump was then energized and left running for approximately 1.5 hours. No abnormalities were observed during this time and the well pump was secured. UFTR Form 0.6A (Unscheduled Reactor Trip Review and Evaluation) was also completed to approve restart. The subsequent restart and operation for over 7 hours on March 8 to complete the A-2 Surveillance was also successful. Based on the indications at the time of the blade drop trip, the well pump 60 amp fuse failed. The fuse failure resulted in a reduction in water flow and an increase in amperage in the other two phases of electrical power. The reduction in water flow caused the "SEC PRESS" trip after a 10 second delay. The increase in amperage of the other two phases caused the overload circuit to trip and remove power to the secondary well pump.

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Number	Date	Description of Occurrence
		All safety and control systems were noted to have operated correctly and in accordance with the Tech Specs for the trip from a known cause. Based on this fact, this blade drop trip is evaluated as not promptly reportable as defined in Technical Specifications Section 6.6.2. This event is also evaluated as having negligible impact on reactor safety or on the health and safety of the public or reactor facility staff. The completed UFTR Form 0.6A (Unscheduled Reactor Trip Review and Evaluation) is Attachment I to the March 2005 monthly report. A memorandum describing the reactor trip event is Attachment II to the March 2005 monthly report.
5.	10 Mar 05	On March 10, 2005, during performance of the quarterly Check of Scram Functions (Q-1 Surveillance), the trip on loss of primary coolant level (item 4 on Page 2 of 6 of the Q-1 Surveillance Data Sheets) was unable to be completed by verifying a change in voltage reading. It was noted that this was a failure of the test mechanism and not a failure of the trip itself. After initial non-invasive investigation of the system, further investigation was made under MLP #05-05 and the decision was made to utilize a test lamp versus a voltage reading and to require that the scram level occur at ≥43.0" versus the previous 44.5" since it could be observed much more accurately with the test lamp installed. A modification package under 10 CFR 50.59 Evaluation and Determination Number 05-01 (Change in Method of Checking PC Level Trip on Quarterly Scram Checks) was prepared, reviewed and approved at a meeting of the RSRS Executive Committee on March 14, 2005 where it was noted that this change is for the method of verifying the scram check for the PC level; there is no change in the scram itself. Essentially, it was agreed that the method change represents an improvement in the method of performing the PC level scram check. Subsequently, after the approval at the RSRS Executive Committee meeting, the procedure change for the Q-1 Surveillance Data Sheets (page 2 of 6, item 4, lines 4–6) was implemented, the changed test method (test lamp) was installed and the PC level trip was verified as required to complete the Q-1 scram checks with no further problems noted.

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(September 2004 - August 2005)

Number	Date	Description of Occurrence
		This failure was noted to be for the test method only and to have had no impact on reactor safety or on the health and safety of the public or the reactor staff. The RSRS Executive Committee meeting minutes without the attachments but including the approval sheet and updated changed Page 2 of the Q-1 Surveillance Data Sheets constitute Attachment IV to the March 2005 monthly report.
6.	27 Jun 05	On June 27, 2005, the reactor was started up at 1553 hours and run at full power starting at 1611 hours using the alternate city water cooling mode to complete the quarterly Radiological Survey of Restricted Areas (Q-5 Surveillance). The Q-5 Surveillance was completed at 1645 hours and the reactor remained at full power to irradiate samples for UCF researchers. Subsequently, a high primary coolant temperature spike at 921.9° F on the SE Fuel Box (Point #1) was received resulting in a blade drop trip at 1653 hours. With all control and safety systems noted to be responding properly, the reactor was secured at 1653 hours. The temperature of 921.9° F on the SE Fuel Box (Point #1) was noted to return to normal again by spiking down at about 1659 hours. Because of the essentially instantaneous >800° F change in temperature on the SE Fuel Box with all other indications normal, this event was evaluated to be due to a failure in the temperature monitoring system rather than to any actual condition requiring a high temperature trip. In particular, the core bulk outlet temperature at Point #8 was noted to be unchanged during the event. Subsequently, the occurrence was evaluated to be due to an intermittent fault in the temperature monitoring system, possibly exacerbated by the ~12° F higher primary coolant exit temperature experienced at full power in the city water cooling mode. Under MLP #05-13, opened to eliminate the computer or thermocouple as the source of the problem, a higher temperature was simulated. Basically, the use of a simulated high temperature (147° F) with no trip proved that the computer/virtual display device works properly. Next, to identify the thermocouple as failed or failing, resistance readings were made on the thermocouple lines for the SE Fuel Box as well as for all other nine (9) points monitored. The results show that, if the SE Fuel Box thermocouple did separate, or any connection to the thermocouple separated, then it has restored itself mechanically. This result indicates the fault is intermittent and difficult to isolate. At this point, UFTR

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(September 2004 - August 2005)

Number	Date	Description of Occurrence
		Form SOP-0.6A (Unscheduled Reactor Trip and Evaluation) was completed, MLP #05-13 was closed out, and the reactor approved for restart subject to a successful daily checkout. Since this occurrence was from a known cause, this trip is not considered promptly reportable, though an NRC Project Manager was told of the occurrence on June 29, 2005 in relation to other issues. In addition, it is evaluated as having had negligible effect on reactor safety and no effect on the health and safety of the public with a successful daily preoperational check completed on June 30, 2005 with no further problems noted at month's end. Completed Form SOP-0.6A (Unscheduled Reactor Trip and Evaluation) is Attachment I to the June 2005 monthly report.
7.	22 Jul 05	Following the high temperature trip (spurious) caused by indicated high temperature on the southeast fuel box outlet thermocouple indication which occurred after operating for 42 minutes at 100 kW (72.333kWh) on June 27, 2005, the event recurred after operating at 100 kW for 27 minutes (47.333 kWh) on July 22, 2005. On July 22, 2005, the reactor was started up at 1555 hours and run at full power starting at 1616 hours using the alternate city water cooling mode to complete an update of part of the quarterly Radiological Survey of Restricted Areas (Q-5 Surveillance) following rearrangement of the rabbit system shielding to reduce radiation levels. The reactor was simultaneously being run to conclude irradiation of samples for UCF researchers. The partial Q-5 Surveillance was essentially completed at 1643 hours when a high primary coolant temperature spike at 921.9° F on the SE Fuel Box (Point #1) was received resulting in a blade drop trip at 1643 hours. With all control and safety systems noted to be responding properly, the reactor was secured at 1644 hours. The temperature of 921.9° F on the SE Fuel Box (Point #1) was noted to return to normal again by spiking down at about 1649 hours. Because of the essentially instantaneous >800° F change in temperature on the SE Fuel Box with all other indications normal, this event was again evaluated to be due to a failure in the temperature monitoring system rather than to any actual condition requiring a high temperature trip, especially since the core bulk outlet temperature at Point #8 was noted to be unchanged during the event. Subsequently, the occurrence was again evaluated to be due

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(September 2004 - August 2005)

Number	Date	Description of Occurrence
		to an intermittent fault in the temperature monitoring system, possibly exacerbated by the ~12° F higher primary coolant exit temperature experienced at full power in the city water cooling mode. The use of a simulated high temperature (147° F) with no trip proved that the computer/virtual display device works properly following the earlier trip. Following the earlier trip, resistance readings were made on the thermocouple lines for the SE Fuel Box as well as for all other nine (9) points monitored. The results showed that, if the SE Fuel Box thermocouple did separate, or any connection to the thermocouple separated, then it has restored itself mechanically. These results had been taken to indicate that the fault is intermittent and difficult to isolate. Nevertheless, it was noted that the system had returned to normal. Since this occurrence was from a known cause, this trip is not considered promptly reportable, though the NRC Project Manager was told of the occurrence on July 22, 2005 because an instant SRO examination had been scheduled for July 25–26, 2005. In addition, it was evaluated as having had negligible effect on reactor safety and no effect on the health and safety of the public as with the previous occurrence. Therefore, it was decided to consider limited UFTR operations before undertaking further maintenance efforts. On July 25, 2005, the RSRS Executive Committee met to review UFTR status and consider possible future operations at limited power levels following the recurrence of the high temperature trip on July 22 after first occurring on June 27, 2005 while operating on lower flow city water secondary cooling resulting in somewhat higher core fuel box outlet water temperatures of 110°-115° F due to unavailability/ unreliability of the secondary well water cooling mode. It was noted that these temperatures are well within the allowable operating temperatures for the cooling water. In both cases the SE fuel box was the indicated failure, though not the hottest box, with the temperature instantaneously spiking to indicate 921.9° F and returning to normal in 5–6 minutes. Because the system had returned to normal and remained so, the RSRS Executive Committee was convened where it was requested that several

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(September 2004 - August 2005)

Number	Date	Description of Occurrence
		lower power operations be approved (providing all indications remain satisfactory) with power levels limited to 25 kW to preclude reaching the temperature where the trip might be initiated.
		The Executive Committee evaluated various input information including the possibility that the high temperature trip is the result of a failing thermocouple or electrical connection and clearly is not an actual temperature transient; that all safety systems have responded properly for both trips; that the likelihood is the intermittent failure will recur at increasing frequency; that the intermittent failure may be due to thermocouple/electrical connections in core and is exacerbated by increased temperature; that the approval for further reduced power operations would be limited in time; that there are multiple indicators to provide reliable reactor core status even after a thermocouple/ temperature monitoring channel failure; that reactor operations are needed for an SRO-trainee's NRC license examination; and finally, that there may be a need for two operators to facilitate completion of surveillances after maintenance to repair the temperature monitoring channel should a core entry be needed.
		After extensive discussions, the Executive Committee voted to approve operations to 25 kW but for only one usage for the SRO-trainee's licensing examination (understood to include prior successful performance of the requisite weekly and daily preoperational checkouts) with no further critical/startup operations allowed until further repairs would be implemented. One committee member also recommended eliminating all possible sources of the trip problem external to the core before undertaking the work to unstack the core.
		The Facility Director was also to discuss this approval with the NRC Project Manager and the NRC Inspector to assure that they were not opposed to this limited operation. This was accomplished effectively in two conversations with the NRC Project Manager who also contacted NRC License Examiner to assure the license examination for the SRO-trainee was rescheduled for August 1–2, 2005.
		Following successful completion of the weekly preoperational checkout on July 25, 2005, and the RSRS Executive Committee agreement to a

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(September 2004 - August 2005)

Number	Date	Description of Occurrence
		single low power run, UFTR Form SOP-0.6A (Unscheduled Reactor Trip Review and Evaluation) was completed and the reactor approved for the low power operation subject to a successful daily preoperational check which was delayed until July 28, 2005 due to an unrelated period trip bistable card failure (see Section II.C.2 of the July 2005 monthly report). The completed UFTR Form SOP-0.6A is Attachment II to the July 2005 monthly report. The Minutes of the RSRS Executive Committee meeting at which the limited restart was approved are Attachment III to the July 2005 monthly report. At the end of July, the reactor had successful weekly and daily preoperational checks completed and was prepared for the low power operation for the license examination which was completed on August 2, 2005. Subsequently, under Maintenance Log Page #05-16, opened after the second trip, and using the half splitting technique, the panel which contains wiring for the thermocouple system was identified in the equipment pit. This panel additionally contains main power lines (but no distribution blocks) for all pit equipment. This arrangement was considered a possible cause of electromagnetic interference and a possible source of the problem with the high temperature indication on the southeast fuel box outlet thermocouple, but it was considered unlikely as the system has operated for a number of years without a problem of the type seen with this high temperature indication. The six in-core thermocouples are all connected to a terminal block within the panel in the pit. These are the only connections on the block. This block has 2 connectors per side, per thermocouple, with 2 sides and therefore contains 12 total connections. Half of these are constantan and the other half copper. This is consistent with Type T (copper-constantan) thermocouple wiring considerations. Under MLP #05-16, mechanical agitation of the system was begun on August 2, 2005 with light initial agitation defined as slowly moving the individual cables as lightly as possible without interfering with the other cables. The proximity of the incoming (from console) and outgoing (to core) lines prevented perfect isolation of the individual lines. Moving one line could (and in some cases did) contact the other lines regardless of the care with which the chore was performed. The

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(September 2004 - August 2005)

Number	Date	Description of Occurrence
		result of the agitation was to generate identical and/or similar temperature spikes on several of the temperature monitoring points including the southeast fuel box outlet (Point #3) and even to a lesser extent on the secondary outlet temperature (Point #10). These results were obtained over the course of several hours, and in some cases were very repeatable. All signal anomalies occurred more than one time. Some anomalies were easily repeated with the same wire motion, others were not. These anomalies were representative of the initial fault which was seen on the southeast fuel box outlet including a spike up and resultant but delayed return to normal. The secondary outlet could not be duplicated. Commonalities considered include the type and age of wire, connection types, connection locations, and path to that location. A general concern at this time was a faulty connection at the block or failure in the wire (hairline fracture) possibly from manufacturing. Subsequently, on August 2, 2005, all connections were reseated (detached and reattached) on the connection block. On August 3, 2005, using a heat gun, the junction box and surrounding wires were heated for 30 minutes. No obvious change occurred during that period, but concern for the insulation necessitated removal of the heat source. A more gradual heat source (1500 W max space heater) was identified and this was used to heat and maintain heat to the area for approximately 2 hours at nearly 90 °F. This is not as hot as the system would probably get during full power operations, especially using city water secondary cooling as was being used at the time of both trips on June 27 and July 22, 2005. During this period no obvious signal changes occurred. Only with mechanical agitation could any changes be induced and these were much reduced, apparently due to the reseating on the connection block completed on August 2, 2005. A substandard mechanism of attachment for several of the spade terminal lugs-to-wire connections was found. The wires were attached by mechanical means employing only a hook through a hole which does not provide a permanent electrical connection. It is possible that the wire, with little heating, could be moving out of contact with the spade

Number	Date	Description of Occurrence
		lug. It was recommended that these junctions be soldered and the system retested. Zinc chloride flux or equivalent was required to solder the constantan joints. At this point the RSRS Executive Committee was convened again, on August 9, 2005. They were requested, upon completion of the soldering and successful retesting, to approve reactor operation at full power with either form of secondary water cooling to verify successful maintenance. After extensive discussions, the Executive Committee voted to approve the 100 kW test run. The resoldering operation was finally completed on August 17, 2005. In addition, thermocouple wire was located and ordered but no replacement thermocouples were located. After August 18, 2005, maintenance time was spent addressing the period bistable failure (see Section II.C.2). The completed UFTR Form SOP-0.6A for the July 22, 2005 trip is Attachment I to the July 2005 monthly report. The Minutes of the August 9, 2005 RSRS Executive Committee meeting at which the limited restart following determination of temperature monitoring wires was approved constitute Attachment I to the August 2005 monthly report which includes the memorandum which outlined the proposed maintenance and subsequent reactor operation to verify operability. At month's end, the reactor was nearly ready for the power run to check repair of the thermocouple monitoring point connections.
8.	25 Jul 05	During performance of the daily preoperational check on July 25, 2005, the 3-second period trip bulb was discovered to be burned out. Upon replacement, the new bulb was found also to be not indicating. Under MLP #05-15, the K5 (period scram) relay was checked visually and it was noted to cycle properly when console power was removed. It was also noted that a 3-second signal gives a blade trip but without proper indication—no period scram light. On July 28, the test adapter was inserted on the A11 card (period bistable card) and the bistable trip was adjusted for audible relay action with no response. The op-amp was then tested for proper switching against the input signal but did not appear to switch apparently because the console switch needed to be reset for proper operation. Using the console key, multiple tests were performed and the K5 card was temporarily switched with a spare with no change. With the A11 card on the extender for evaluation, the

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Number	Date	Description of Occurrence
		period scram relay was noted to be operating properly, possibly due to mechanical agitation; however, with the extender card removed the problem persisted. Next the K5 relay was temporarily switched with a spare since the indicating bulb was failed initially so arcing at the relay points was possible. The blade continued to trip on a 3-second period but without proper indication (no period scram light). Subsequently, the K5 and K4 relays were swapped for testing but again the problem remained so relays were replaced to original locations. At this point, the problem was isolated to an apparent failure on the A11 card. Replacements were then ordered for the HP11D relay on the card; no replacements were identified for the card's HGSM5001 wetted relay. On July 28, 2005, the HP11D relay was removed and replaced with the newly acquired spare. The A11 card was then replaced and the period trip test performed 4 times successfully. The water level was then raised and Section 7.2.5.5.2 of the daily preoperational checkout (period trip test) was again performed 4 times with all tests satisfactory so the period bistable trip failure was considered repaired. Since this failure of the period bistable trip was discovered during preoperational checks, it is not considered promptly reportable. This event is considered to have had no impact on the health and safety of the public or facility personnel and negligible impact on reactor safety. During a subsequent daily checkout on August 18, 2005, the period trip test failed again. MLP #05-15 was reopened on August 18 for troubleshooting and analysis of the intermittently failing period bistable trip. After some testing it was discovered that the semi-conductor signal chain in the A11 bistable card was at fault. Two of the semi-conductors were difficult to obtain (one of which still has not been obtained, the A51 operational amplifier). At this point it was deemed that testing should continue with the available part, so the Q51 equivalent transistor was subsequently replaced on August 19, 2005 under 10 CFR 50.59 Evaluation Number 05-03 (Transistor Change in A11 Board for Failed Period Trip Bistable). After installation, the period bistable trip was checked satisfactorily three times but then it failed two times. It was thought the failures might be occurring due to temperature effects, so several temperature tests were performed on August 22, but these tests were unsuccessful or successful whether at

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Number	Date	Description of Occurrence
		lower temperatures soon after restoring electrical power or after power was restored for a longer period. Though not certain, the apparent failure was thought to be the operational amplifier on the A11 board. Testing showed that the replaced transistor was not at fault but it did provide isolation for the other transistor leaving only the operational amplifier as the failed component. However, as of August 22, no source could be identified for the operational amplifier that could be counted on. The only source found was Dionese Electronics in Argentina which was uncertain. After some further discussion with General Atomics, it was discovered that this card in a later revision was available for ~$3,000. General Atomics sent the card priority overnight on August 23 and the card arrived on August 24, 2005 when it was tested to assure basic operation and compatibility. It was noted that the version of the NT-4 card used at the UFTR requires a 56.2K 1% 1 Watt precision resistor. This resistor could not be readily obtained on the open market so General Atomics finally provided one through Mouser Electronics but it proved to be the wrong wattage.
		Caddock Electronics was then contacted on August 26, 2005 and agreed to make the substitute resistor at no cost but since it was being manufactured, it would not be available until August 30. On August 30, 2005 the precision resistor was received (along with spares). The resistor was then installed on the A11 card under 10 CFR 50.59 Evaluation Number 05-04 (Replacement/Upgrade of NT-4 Bistable Card for Period Scram/Fast Period Interlock (A11 Card)) on August 31, 2005 and a series of tests were run to assure proper functioning of the 10 second fast period interlock and the 3 second period scram. These values were aligned and tested satisfactorily with no fewer than 16 tests in both the water-down and water-up/blade-up configurations. All blades dropped as expected and a successful daily checkout was performed to assure the bistable period trip was operating properly with no further problems noted.
		Since this failure of the period bistable trip was discovered during preoperational checks, it is not considered promptly reportable. This event is considered to have had no impact on the health and safety of the public or facility personnel and negligible impact on reactor safety.

IV. MODIFICATIONS TO THE OPERATING CHARACTERISTICS OR CAPABILITIES OF THE UFTR

A number of modifications and/or changes in conditions were made to the operating characteristics or capabilities of the UFTR and directly related facilities during the 2004–5 reporting period. These modifications and/or changes in conditions were all subjected to 10 CFR 50.59 evaluations and then determinations (as necessary) to assure that no unreviewed safety questions were involved.

- <u>Carried over from the 1984–85 Reporting Year</u>:

  Modification 7: Addition of Secondary Water Flow Sensors (Rotameters)

- <u>Carried over from the 1991–92 Reporting Year</u>:

  Modification 92-04: Installation of New Manometers on Core Vent System

  Modification 92-06: Modification to the UFTR Thermocouple System: Implementation of Terminal Strips and Quick Disconnects

- <u>Carried over from the 1996–97 Reporting Year</u>:

  Modification 96-13: Security System Power Pack Replacement

- <u>Carried over from the 2003–4 Reporting Year</u>:

  Modification 04-01: Modification/Upgrade of Chilled Water System for Reactor Building – Pipe Supports/Wall Anchors

1. <u>Security System Power Pack Replacement (Permanent – Open Item)</u>

(Modification 96-13: Evaluation Completed December 1996)
(Modification 99-02: Evaluation Completed 11 February 1999)

Following one spurious security alarm on November 10 and two alarms on November 11, 1996, the security system batteries were checked and replaced (S-7 Surveillance). Under MLP #96-30 the rechargeable batteries were found to be low and were recharged. Subsequently, 10 CFR 50.59 Evaluation Number 96-13 was developed to allow modification and replacement of the power pack to prevent recurrence of the problem of spurious alarms due to low voltage. Measurements were made and security system circuits checked and verified. In addition, the 6 volt batteries were recharged in mid-month. At the end of November 1996, the design and development of a new power pack per 10 CFR 50.59 Evaluation Number 96-13 was in progress; at the end of December 1996, the 10 CFR 50.59 Evaluation is complete as is the design, with installation of the new power supply on January 7, 1997 with all but one siren operational to meet requirements. Subsequently, the west lot siren was repaired on January 13 and both the west lot and journalism side siren horn drivers wiring was reterminated on January 14, 1997. Drawings and maintenance log were subsequently updated and an evaluation made that separate grounds would be needed for the security system batteries to assure proper charging and eliminate spurious alarms as the batteries discharge over time. On March 10, 1997, the power supply was removed for modification. Upon installation, various problems occurred resulting in partial and intermittent compensated outage of the security system over the period March 10-21 with circuit mapping performed for troubleshooting on March 19 and the intermittent ground finally repaired on March 21, 1997, but without installation of the modification to separate grounds, basically returning the system to its state prior to March 10. Subsequently, the 4 volt rechargeable batteries have been replaced on May 14, June 18, July 7, and July 24, 1997 (for prevention purposes on July 30, 1997), on August 29, and on September 29, 1997. Following a full S-7 Surveillance on October 24, 1997, the loss of the holdup alarm was corrected under MLP #96-30 by reterminating a loose wire. Subsequently, the 4 volt rechargeable batteries were replaced on December 16, 1997 and again on January 9, February 10, March 10, April 8, and on May 6, 1998. Following a full S-7 Surveillance on May 27, 1998, the 4 volt rechargeable batteries were replaced again on June 24, July 24, August 19, September 16 and October 13, 1998. Following a full S-7 Surveillance including replacement of rechargeable batteries on November 10, the 4 volt rechargeable batteries were replaced again on December 7, 1998 and January 4, February 1 and March 2, 1999 with upgraded 4 volt batteries installed on March 12, 1999 under 10 CFR 50.59 Evaluation Number 99-02 developed and approved in February to upgrade the 4 volt rechargeable batteries for longer life. There had been no need for further replacement through the end of July 1999 though the full S-7 Surveillance was performed on July 2, 1999. Following the full S-7 Surveillance, when the 4 volt batteries were not replaced, the 4 volt rechargeable batteries were replaced again on August 24, 1999. The 4 volt rechargeable batteries were replaced again on February 24, 2000. There had been no further need for replacement until completion of the full S-7 Surveillance on May 25, 2000. The 4 volt rechargeable batteries were again replaced on November 10, 2000 followed by a full S-7 Surveillance on December 29, 2000. The 4 volt rechargeable batteries were replaced again on February 26,

2001. There had been no further need for replacement until completion of the full S-7 Surveillance on May 22, 2001. Subsequently the 4 volt rechargeable batteries were replaced again on August 24 and on November 9, 2001 while a full S-7 Surveillance without replacement of the 4 volt batteries was conducted on December 3, 2001. Subsequently, the 4 volt rechargeable batteries were replaced again on January 16 and on March 29, 2002 while a full S-7 Surveillance was conducted on June 6, 2002. Subsequently, the holdup alarms' batteries were replaced due to low voltage on August 16, 2002 and the 4 volt rechargeable batteries were replaced again on August 21, 2002. Current plans are to replace the entire system with an equivalent one with DOE 2001–2 URI grant funds. A full S-7 Surveillance was conducted on October 28/31, 2002. Subsequently, the 4 volt rechargeable batteries were replaced again on January 2 and on March 11, 2003, with another full S-7 Surveillance conducted on April 25, 2003. Subsequently, the 4 volt rechargeable batteries were replaced again on June 11 and on August 26, 2003. There had been no further need for replacement until completion of the full S-7 Surveillance on November 7, 2003. Subsequently, the 4 volt rechargeable batteries were replaced again on December 11, 2003.

The 4 volt rechargeable batteries were replaced again on January 22, 2004 with another full S-7 Surveillance conducted on April 7, 2004. Subsequently, the 4 volt rechargeable batteries were replaced again on May 6, 2004 and on July 14, 2004 with another full S-7 Surveillance conducted on October 15, 2004 without replacement of the 4 volt rechargeable batteries. Subsequently, the 4 volt rechargeable batteries were replaced again on December 16, 2004.

The 4 volt rechargeable batteries were replaced again on February 21 and April 15, 2005 with another full S-7 Surveillance conducted on June 20–21, 2005.

Controlling Documents:	Maintenance Log Page #96-30 (Remains Open)
	10 CFR 50.59 Evaluation Number 96-13
	10 CFR 50.59 Evaluation Number 99-02

2. Modification/Upgrade of Chilled Water System for Reactor Building – Pipe Supports/Wall Anchors (Permanent – Closed Item)

(Modification 04-01 Evaluation Completed 5 August 2004)

For some time, plans had been made to conduct work in the west lot for the Weil Replacement Chill Water project including meetings, discussions and visits for measurements by and for various personnel including UF PPD Project Manager, PPI supervisor, Matt Seales of Perry Construction and foreman Jimbo Williams of WW Gay. Initial work began under MWO #0674958 on August 5, 2004 with temporary movement of the north section of the west lot fence to allow clearing room for pipes under supervision of WW Gay foreman Jimbo Williams. No further work was accomplished inside the fenced area until August 18, 2004. Subsequently, holes were drilled in the reactor cell west wall under 10 CFR 50.59 Evaluation and Determination Number 04-01 (Modification/Upgrade of Chilled Water System for Reactor Building – Pipe Supports/Wall Anchors) to provide anchoring for the chill water line pipe supports on August 19, 2004. The minutes of the

August 19, 2004 meeting of the RSRS Executive Committee which reviewed and approved these anchors is Attachment II to the August 2004 monthly report. With installation of the main chill water pipes below ground leading into the west lot, the west lot fence was restored to its original location in improved condition on August 20, 2004 as Tom Quarles of Florida Enterprise Corporation replaced the barbed wire as well. At year's end the main piping is in place and efforts involving daily access are underway to complete the pipe installation. During September 2004, the chilled water line replacement project continued with the usual vehicular, equipment, supplies and personnel access until September 14 when most work was completed and personnel were removed to another project. Perry Construction supervisor Matt Searles accessed the west lot on September 15, 16 and 22 to perform some grout work with no further work accomplished through the end of September. On October 11, 2004, Troy Lauramore and then Matt Searles contacted the facility concerning access for some work on October 21-22 prior to hook up of the new chill water lines scheduled for October 23, 2004. Matt Searles then arranged for access and Jimbo Williams of WW Gay then accessed the west lot with Richard Ellison and Matthew Kight to make final measurements. On October 19 plans for final hookup on October 23 were verified with Matt Searles and discussions were conducted with the PPD alarm systems supervisor concerning fire alarm cut off requirements. Subsequently, a PPD alarms system technician visited and verified that the chill water final hookup would not require bypassing any portions of the fire alarm system. Subsequently, an university architect and the COE Services supervisor walked through to check on the west lot chiller work and plans for the reactor building annex remodeling. As planned, on October 23, the PPD supervisor, Perry Construction supervisor Matt Searles and WW Gay Foreman Jimbo Williams and various support personnel (William Brown, Matthew Kight, Caleb Smith, Lance Hersey and forklift operator Don Heinz) accessed the west lot with various equipment and performed final hookup installation of the new chill water lines. Subsequently, on October 24, Matt Searles made arrangements for final insulation installation with Gaylord Insulation personnel (Steve Taylor, Arthur Gilbert and Paul Videon) accessing the west lot on October 26 and 27 to perform the work. At the end of October little remained to be accomplished on this project with a substantial completion inspection planned for November. On November 1, 2004, Michael Ellison of WWGay accessed the west lot twice to perform some cosmetic repairs. Finally, on November 10, the substantial completion inspection was accomplished with those present including Matt Webster and Matt Searles of Perry Construction, Jimbo Williams of WW Gay, and Supervisor Mike Vaughn, Senior Engineer Gray Rawls, Project Manager Jeff Bair, and Charles Milford of PPD.

Controlling Documents: Maintenance Work Order #0674958
10 CFR 50.59 Evaluation and Determination Number 04-01

3. <u>Change in Method of Checking PC Level trip on Quarterly Scram Checks (Permanent – Closed Item)</u>

(Modification 05-01 Evaluation Completed 5 August 2004)

On March 10, 2005, during performance of the quarterly Check of Scram Functions (Q-1 Surveillance), the trip on loss of primary coolant level (item 4 on Page 2 of 6 of the Q-1 Surveillance Data Sheets) was unable to be completed by verifying a change in voltage reading. It was noted that this was a failure of the test mechanism and not a failure of the trip itself. After initial non-invasive investigation of the system, further investigation was made under MLP #05-05 and the decision was made to utilize a test lamp versus a voltage reading and to require that the scram level occur at ≥43.0" versus the previous 44.5" since it could be observed much more accurately with the test lamp installed. A modification package under 10 CFR 50.59 Evaluation and Determination Number 05-01 (Change in Method of Checking PC Level Trip on Quarterly Scram Checks) was prepared, reviewed and approved at a meeting of the RSRS Executive Committee on March 14, 2005 where it was noted that this change is for the method of verifying the scram check for the PC level; there is no change in the scram itself. Essentially, it was agreed that the method change represents an improvement in the method of performing the PC level scram check.

Subsequently, after the approval at the RSRS Executive Committee meeting, the procedure change for the Q-1 Surveillance Data Sheets (Page 2 of 6, item 4, lines 4–6) was implemented, the changed test method (test lamp) was installed and the PC level trip was verified as required to complete the Q-1 scram checks with no further problems noted. This failure was noted to be for the test method only and to have had no impact on reactor safety or on the health and safety of the public or the reactor staff. The RSRS Executive Committee meeting minutes without the attachments but including the approval sheet and updated changed Page 2 of the Q-1 Surveillance Data Sheets constitute Attachment IV to the March 2005 monthly report.

Controlling Documents: Maintenance Log Page #05-05
10 CFR 50.59 Evaluation and Determination Number 05-01

4. <u>Upgrade Replacement of UFTR City Water Flow Meter (Permanent – Closed Item)</u>

(Modification 05-02 Evaluation Completed 2 May 2005)

When the deep well pump tripped out due to a blown fuse, it was decided to try operating with the city water cooling system. However, several checks of the city water trip were unsuccessful due to the flow meter valve not closing completely. Subsequently, under MLP #04-29, the actuator on the flow meter valve was lubricated but there was little effect on the operation with no further work attempted by the end of December 2004. In April 2005, a replacement flow meter was identified to avoid failure problems; this meter was ordered on May 2 and finally received on May 9, 2005. Subsequently, plans were made to install the new flow meter with 10 CFR 50.59 Evaluation Number 05-02 (Upgrade Replacement of UFTR City Water Flow Meter) developed to control the installation. All the necessary support materials were assembled and installation of the new flow meter was undertaken beginning on May 24 and continuing to May 25 to correct initial minor leaks and assure proper operation with subsequent completion of documentation with several subsequent verifications of proper flow meter operation with no further problems noted.

Controlling Documents: Maintenance Log Page #04-29
10 CFR 50.59 Evaluation Number 05-02

V. SIGNIFICANT MAINTENANCE, TESTS AND SURVEILLANCES OF UFTR REACTOR SYSTEMS AND FACILITIES

A review of records for the 1984–85 reporting year showed extensive corrective and preventive maintenance was performed on all four control blade drive systems <u>external</u> to the biological shield. Similarly maintenance work during the 1985–86 reporting year was even more extensive as the problem of a sticking safety blade (S-3) recurred on September 3, 1985. The recurrence necessarily demanded a detailed and complete check of all control blade drive systems to determine finally and correct the cause of the sticking blade internal to the biological shield with the 1986–87 reporting year involving relatively little maintenance and no large maintenance projects.

For the 1987–88 reporting year, there were two dominant though manageable maintenance projects. The first large scale maintenance project during the 1987–88 reporting year involved an extensive effort to clean the control blade drive motor gear assemblies to free them of hardened grease and replace worn bearings. The second large-scale project involved the evaluation, corrective action, testing and monitoring of the two safety channels due to two occurrences of the downscale failure of the Safety Channel 1 meter indication (and probably the function). This was the largest maintenance effort since the control blade drive system maintenance performed internal to the biological shield in the 1985–86 reporting year. The 79.2% availability for the 1987–88 year indicated more or less routine maintenance and surveillance checks and tests throughout the year except for the two large projects cited above.

For 1988–89, the availability was up to 87.67%. Of the 45 equivalent full days of unavailability, only 28.25 days were actually due to forced unavailability primarily due to corrective maintenance for repairs. There was no single project dominating unavailability, though multiple maintenance tasks on the two-pen recorder and on the Radiation Monitoring System clearly warranted consideration of replacing these items when funds could be made available.

Maintenance efforts in the 1989–90 reporting year increased again so that total availability for the year was only 68.84%. Especially significant efforts were devoted to checks, repairs, surveillances and other maintenance activities connected with the biennial fuel inspection resulting in a two-month outage, part of which was due to the final failure and subsequent replacement of the two-pen log/linear recorder. Though no other single maintenance effort was really large, there was considerable effort devoted to Safety Channel and other control and reactor protection system-related repairs during the year both for repairs following trips or other failures and for preventive maintenance. Certainly, the 113.75 total days unavailability (31.16% unavailability) was one of the poorer records in recent years.

Although availability in the 1990–91 reporting year was not as high as hoped, it was greatly improved as there were 93 days forced unavailability, 1.25 days planned unavailability and 23.25 days of administrative shutdown for an overall availability of 67.81%. Primary sources of forced outage time were replacement of seals and connectors on the primary coolant system and extensive maintenance performed to complete the nuclear instrumentation calibration. These values were somewhat elevated, especially administrative shutdown time, by the lack of a full-time Reactor

Manager and lack of replacement part inventory along with a shortage of licensed personnel, especially senior reactor operators over the last six months of the year.

Although no permanent Reactor Manager was able to be hired in the 1991–92 reporting year, two new part-time student senior reactor operators (SROs) were licensed and certified on October 17, 1992. Although availability in the 1991–92 reporting year was not as high as had been hoped, availability was again improved significantly as there were only 72.25 days forced unavailability, 4.25 days planned unavailability and 23.50 days of administrative shutdown. The 76.50 days total unavailability (20.90% unavailability) for maintenance is approximately average for the past decade. Again, these values for unavailability were elevated by the lack of a full-time Reactor Manager, especially early in the reporting year before certification of the two new SROs. With the appointment of a part-time Acting Reactor Manager on August 11, 1992, this situation improved in the next reporting year.

Although there were no large maintenance projects for the 1991–92 year, several major projects contributed to forced unavailability. First, and most significantly, two failures of the thermocouple connections to the south center fuel box were responsible for over 31 days of forced unavailability. Similarly, various failures related to the nuclear instrumentation system, including Safety Channel 2 trip indication, Safety Channel 2 meter circuit, Safety Channel 1 +15 volt and high voltage power supplies and the control blade position indicating circuits as well as replacement of bearings and pillow blocks for the stack diluting fan and the motor on the deep well pump were responsible for significant amounts of forced unavailability. As is indicated, these four areas account for most of the forced unavailability for the 1991–92 reporting year with the failed thermocouple connections and the safety channels meriting the most concern for preventive maintenance.

Although a permanent Reactor Manager was not hired until July 1993, the availability of part-time operators was good throughout the 1992–93 reporting year. Availability in the 1992–93 reporting year returned to a high level as there were only 22.63 days forced unavailability, 12.63 days planned unavailability and 11.50 days of administrative shutdown for a total of 46.75 days unavailability and an overall availability of 87.23%. The 35.25 days total unavailability (9.66% unavailability) for maintenance was one of the best in ten years. With appointment of a full-time Reactor Manager in July 1993 it was hoped this situation could be improved even further in the next year though much would depend on support for part-time personnel. Significant sources of forced unavailability for the 1992–93 reporting year were repair of deep well pump piping, adjustment and repair of Safety Channel 1 during the annual calibration and repair of the north side core area thermocouple connections and replacement of wiring following failure of temperature point #4 plus repeated small outages and several unscheduled shutdowns due to failures of the control blade position indicators/indicator circuits with an effort planned to replace these nixie tube systems in the next reporting year.

With a full-time Reactor Manager available for the full 1993–94 reporting year, good availability of other licensed and unlicensed personnel and no large maintenance efforts, availability for the 1993–94 reporting year was even better than in the previous year. There were only 21.38 days forced unavailability; 13.25 days planned unavailability and 3 days of administrative shutdown for a total of 37.625 days unavailability and an overall availability of 89.69%. Significant sources of forced unavailability were to check out and verify proper detector current and operation of the compensated ion chamber and linear (red) pen following failure due to excessive moisture in October 1993, to check, locate and correct erratic response in the Safety-3 control blade position indicating (BPI) circuit in December 1993 and January 1994, to locate and correct an open circuit in the Safety-3 control blade drive circuit in January/February 1994, and to replace the intermittently failing shield tank water level trip magnetic reed switch in February 1994. The replacement of the nixie tube indicators in the control blade position indicating circuits in June 1994 promised to reduce forced outages from failures of the BPI circuits in the future.

With a full-time Reactor Manager again available for the full 1994–95 reporting year, reasonable availability of other licensed and unlicensed personnel and a limited number (3) of medium length forced outages, availability for the 1994–95 reporting year was only slightly reduced to 88.15% from the previous year. There were 26.50 days forced unavailability, 11.75 days planned unavailability and 5 days administrative shutdown. The three significant sources of forced unavailability were for the outage to address the anomalous primary coolant resistivity drop in March 1995, for the outage to remove debris and perform checks of the primary coolant system return line flow trip switch following removal of debris in June 1995, and finally for the outage to repair the automatic flux controller in August 1995 and which was still in progress at year's end.

With a full-time Reactor Manager again available for most of the 1995–96 reporting year, limited somewhat by family illness until resigning the position effective August 9, 1996, and with reasonable availability of other licensed and unlicensed personnel, but with several (3) medium length forced outages plus considerable planned outage time for roof repair, availability for the 1995-96 reporting year was somewhat reduced to 75.68% from the previous year. There were 44.875 days forced unavailability, 41.875 days planned unavailability and 2.25 days administrative shutdown for a total unavailability of 89 days. The three significant sources of forced unavailability were for the continued outage at the beginning of the year in September 1995 for the outage to repair the automatic flux controller begun in August 1995, for the outage to repair the linear (red) pen circuit in October 1995, and for the outage to troubleshoot and repair the Safety Channel 2 loss of high voltage monitoring circuit in April 1996 and again in July 1996. There was also significant planned outage time for the year for two surveillances to complete the inspection of mechanical integrity of the control blade drive systems internal to the biological shielding (V-1 Surveillance) in December 1995 and the biennial inspection of incore fuel elements (B-2 Surveillance) in August 1996. Similarly, the contract work to replace and then repair the reactor building roof involved considerable planned unavailability throughout the 1995–96 year and was still in progress at the end of the 1995–96 year.

With a full-time Reactor Manager only available for about three months beginning in late December 1996 until March 28, 1997, plus the loss of one part-time SRO and the licensing of another in midyear leading to somewhat restricted availability of licensed as well as unlicensed personnel, plus considerable forced outage time for replacement of failed equipment and some planned outage time for conducting and improving the annual calibration checks of nuclear instrumentation, availability for the 1996–97 reporting year was further reduced to 62.20% from 75.68% the previous year. There were 102.25 days forced unavailability, only 16.625 days planned unavailability and 4.50 days administrative shutdown. The three most significant sources of forced unavailability were for the outage to replace the failed compensated ionization chamber (CIC) with the uncompensated ionization chamber (UIC) run in CIC mode, to obtain a new UIC, to replace the connectors and cables on both detectors and then test and assure proper calibration of the nuclear instruments in September to December 1996 (72.875 days); for replacement of the shield tank demineralizer system pump including flow circuit rearrangement in July/August 1997 (20.875 days); and replacement of a failed reed switch in the primary coolant level trip circuit in July 1997 (2.75 days). There was also significant planned outage time for the year to make adjustments and rework the annual calibration of nuclear instrumentation (A-2 Surveillance) in March 1997 (10 days) plus continuing periodic contract work to replace and then repair/upgrade the reactor building roof until June 1997 (4.75 days).

With a full-time Reactor Manager not available at all for the 1997–98 reporting year plus the extended outage beginning in May 1998, the hiring of two SRO-trainees did not result in the licensing of any new operators for the 1997–98 year resulting in continued somewhat restricted availability of licensed as well as unlicensed personnel, plus considerable forced outage time— some involving failed equipment but the vast majority to investigate the cause of the reactivity anomaly resulting in higher than expected critical regulating blade position. There was also some planned outage time, mostly for conducting and improving the annual calibration checks of nuclear instrumentation. Therefore, availability for the 1997–98 reporting year was further reduced to 58.29% from 62.20% the previous year. There were 131.375 days forced unavailability, only 13.375 days planned unavailability and 7.50 days administrative shutdown. The most significant source of "forced" unavailability was the outage to investigate the reactivity anomaly lasting from the beginning of May through the end of the year in August (122.25 days). Only two other sources of forced outage time accounted for over two days; repair of the failure of the Safety Channel 2 high voltage power supply loss of high voltage trip (2.875 days) and replacement of a failed reed switch on the primary coolant return line flow sensor (2.875 days), both in April 1998. Several pieces of maintenance would have involved significant forced outage in the last few months of the year except the reactor was already unavailable due to addressing the reactivity anomaly. There was also significant planned outage time for the year to make adjustments and perform the annual calibration of nuclear instrumentation (A-2 Surveillance) in March 1998 (10.75 days).

With no full-time Reactor Manager for the entire1998–99 reporting year plus the outage for the reactivity anomaly extending until return to normal operations on August 17 (regular operations began on August 9 but delayed operations training had to be conducted), neither of the two SRO-trainees was able to be licensed with most of the year's outage attributed to addressing the reactivity anomaly and returning the UFTR to normal operating status after completing all required surveillances as well as delayed annual reactor operations tests. Therefore, availability for the 1998–99 reporting year was further reduced to only 4.01% from 58.29% in the previous year. Basically, there were 348.625 days forced unavailability, 0.375 day planned unavailability (in August 1999) and no days administrative shutdown as such. Of course, this forced unavailability was essentially all to address investigation of the reactivity anomaly though a number of other events during the year could have impacted unavailability had the reactor been in an operational status.

With a 90% full-time Acting Reactor Manager for the entire 1999–2000 reporting year and successful recovery from the outage to address the reactivity anomaly for most of the previous year plus licensing of a new senior reactor operator from February 15, 2000 through the end of the reporting year, availability was restored to relatively high levels. Availability for the 1999–2000 reporting year was increased to 88.19% from 4.01% in the previous year. Basically, there were

20.875 days forced unavailability, 14.50 days planned unavailability and 8.25 days administrative shutdown. The forced unavailability was primarily due to repairs on the failed temperature monitor (11 days in October and 1.25 days in June) plus repair of the failed auxiliary stack monitor meter/alarm (2.875 days), repair of the failed green pen mount on the two-pen recorder (1.125 days) and replacement/cleaning and reseating relays to address failure of the dump valve to close. The only significant planned outages for the 1999–2000 reporting year were to replace/upgrade overhead lighting in the cell/control room (3.50 days) and then to make adjustments and perform the annual calibration of nuclear instrumentation (A-2 Surveillance).

With a 90% full-time Acting Reactor Manager again for the 2000–2001 reporting year, availability of personnel was maintained during the year though one half-time SRO resigned for a well-paying industry position in December 2000. The various outages for the year made it difficult to train new operators so no new operators were licensed during the year. However, with one five-eighths-time operator-trainee available for the whole year and another available from mid-January 2001 to the end of the year, personnel availability was good. Unfortunately, forced outages presented a problem. Availability for the 2000–2001 reporting year was decreased to 58.47% from 88.19% in the previous year. Basically, there were 128.625 days forced unavailability, 15.25 days planned unavailability and 7 days administrative unavailability. The large number of days of forced unavailability was primarily due to a series of equipment failures for a broken primary coolant rupture disk (3.875 days in September 2000), repair of the solenoid on the PC dump valve (10.25 days in October 2000), replacement of a failed two-pen recorder (12 days in January 2001), repair and eventual replacement of failed temperature monitor/recorder with computer-based system (61.875 days in January–April 2001), and troubleshooting to evaluate and repair failed wide range drawer (36 days in July–August 2001) extending into the next reporting year. The only significant planned outage for the year was to make adjustments and perform the annual calibration of nuclear instrumentation (A-2 Surveillance) (12 days in January and April 2001) spread out due to two-pen recorder and temperature monitor/recorder failures.

For the 2001–2 reporting year a two-thirds time SRO/Acting Reactor Manager was available for three months of the year to aid in recovery from the outage to address future the wide range drawer which was completed in mid-October 2001 accounting for 45.25 forced outage days. Subsequently there was high availability and usage for four months. However, with reduction to one-quarter time for three months for the SRO/Acting Reactor Manager, and then termination at the end of February 2002, the facility was left with only one licensed SRO for the last half of the reporting year. The facility was then subjected to a number of failures, the most serious was the failure of the fission chamber—the outage occupied 169.375 days through the end of the reporting year. Other significant outages were for a broken ruptured disk (6.125 days) in December 2001/ January 2002 plus an 8-day "planned" outage to repair scram annunciator light bulb holder and spacer clips in July 2002. The result was an availability of only 34.2% for the 2001–2 reporting year.

For the 2002–3 reporting year there was no reactor manager with one part time SRO plus the Director to start the year to address the failed fission chamber extending over the first 192.375 days of the reporting year. The part time SRO resigned effective at the end of April 2003 with two more part time student SROs licensed in late May 2003. Subsequent to the fission chamber outage availability was relatively high though outages for a failed deep well pump (8.375 days) and for a failure of the S-2 control blade to drop (12.375 days) contributed to nearly 232 days unavailability for the year and annual availability was attributable to limited licensed staff especially until two more part time student SROs were licensed in late May 2003. Interestingly enough the availability for the final few months of the reporting year was over 91% and the potential outage for a sticky control blade lasted on 12.375 days in June. Nevertheless, the resultant yearly average availability for the 2002–3 reporting year was only slightly better than the previous year at 36.5% versus 34.2%.

For the 2003–4 reporting year there was no reactor manager with one part time SRO who served occasionally as Acting Manager in the Director's absence plus the Director for the entire year. This part time SRO resigned after graduation in April 2004 and effective May 28, 2004 but continued to be employed for the remainder of the reporting year. After the extended forced outage rate in the previous two reporting years, the 2003–4 reporting year saw a return to relatively high availability with only three forced outages exceeding one (1) day including 3.375 days in October 2003 to repair the secondary cooling flow meter, 5.625 days in April 2004 to repair the log channel on the two pen recorder and 14.375 days in July–August 2004 to correct the problem of a sticking S-2 control blade. The only other equipment-related lengthy outage was for 10.625 days planned unavailability to make adjustments and perform the annual calibration of nuclear instruments (A-2 Surveillance). The 2003–4 overall availability was at 85.01% with overall unavailability at 14.99% (54.875 days) with only 26.375 days forced unavailability, 13 days planned unavailability and a relatively high administrative unavailability of 15.50 days primarily for vacations and holidays. Certainly the 85.01% availability in the 2003–4 reporting year is far better than the 36.5% in the 2002–3 reporting year or the 34.2% in the 2001–2 reporting year.

For the 2004–5 reporting year there was again no reactor manager with a part time SRO who served occasionally as Acting Manager in the Director's absence until his departure on March 22, 2005. The 2004–5 saw a maintenance of relatively high availability with less than 43 total days unavailability including 21.875 days administratively unavailability for the first ten months of the year. There were five forced outages exceeding three days for the reporting year with all but one occurring prior to late June 2005 including 4 days in March 2005 to design, install, check and implement a new means to check the primary coolant level trip, 8 days in July to clean and repair the secondary cooling flow meter and well water flow trip, 16.875 days in July/August 2005 to repair the period trip bistable card, 18.125 days (primary) in June–August 2005 to address trips on high primary coolant southeast fuel box outlet temperature and 3.375 days in August 2005 to address a break in the secondary heat exchanger sample line. The only other equipment related lengthy outage was 6.875 days planned unavailability in March 2005 to make adjustments and perform the annual calibration of nuclear instruments (A-2 Surveillance). The 2004–5 overall availability was at 73.77% with overall unavailability at 26.23% (95.75 days) with 49.50 days forced outage unavailability, 11.375 days planned unavailability and a high administrative unavailability of 34.875 days primarily for vacations, meetings, holidays and hurricane watches. The 73.77% availability in the 2004–5 reporting year is somewhat below the long term average for the UFTR but much better than the 36.5% in the 2002–3 reporting year or the 34.2% in the 2001–2 reporting year.

In the tables that follow, all significant maintenance, tests and surveillances of UFTR reactor systems and facilities are tabulated and briefly described in chronological order; these tabulations also include administrative checks. Table V-1 contains all regularly scheduled surveillances, tests or other checks and maintenance required by the Technical Specifications, NRC commitments, UFTR Standard Operating Procedures, or other administrative controls; these items are normally delineated with a prefix letter and a number for tracking purposes. The number of these surveillances increases each year as the UFTR QA Program matures and requirements become more restrictive.

A listing of all the maintenance projects required to repair a failed system or component or to prevent a failure of a degraded system or component is presented in Table V-2. These maintenance efforts are frequently not scheduled though they can be when a problem is noted to be developing and preventive actions are implemented. In addition, they frequently are associated with reactor unavailability. Finally, these maintenance items can be associated with surveillances, checks or test items listed in Table V-1 since some of these scheduled surveillances are also required to be performed on a system after the system undergoes maintenance. For example, when the area monitor check sources or detectors are the subject of preventive or corrective maintenance as listed in Table V-2, the Q-2 calibration check of the area monitors must be completed as listed in Table V-1 before the reactor is considered operable. Similarly, when maintenance is performed on the control system, various surveillances such as control blade drive time and drop time measurements must be performed satisfactorily before the reactor can return to normal operations.

In Table V-2 the first date for each entry is the date when the Maintenance Log Page (MLP) was opened; in quite a few cases, this date may be one or more days after the original problem was noted. The date for work completion and the MLP number are included at the end of the maintenance description. As a result, in some years the first items listed in Table V-2 can have a starting date prior to the beginning of the current reporting year as the maintenance could be completed in a subsequent reporting year. This is the case for the first six entries in Table V-2 which involved maintenance in progress at the end of the 2003–4 reporting year; indeed the first item was opened during the 1993–94 reporting year as MLP #94-14 used to control planned installation of a new area radiation monitoring system. The second of the three entries (MLP #96-30 to control repair and upgrade of the security system) was opened during the 1996–97 reporting year while the third entry (MLP #02-26 to control repair of nimbin modules) was opened during the 2002–3 reporting year. The other three items were all carried over from August 2004 and closed out early in the reporting year. The other three items were all carried over from August 2004 and closed out early in the reporting year.

Similarly, seven Maintenance Log Pages remain open at the end of the current 2004–5 reporting year. MLP #94-14 to control installation of a new area radiation monitoring system, MLP #96-30 to control repair and upgrade of the security system, and MLP #02-26 to address repair of a portable nimbin single channel analyzer and timer/counter modules all remain open from the previous year. It is expected that MLP #94-14, MLP #96-30 and MLP #02-26 will be open for some time as implementation of the new area radiation monitoring system is a major modification, upgrade of the security system will be time consuming and expensive and repair of the nimbin modules requires specialized expertise and will be expensive so it is not a high priority. The other three maintenance items remaining open include MLP # 05-14 to address repairs of the secondary cooling well water flow meter, MLP #05-16 to address thermocouple or connecting wiring intermittent failure for temperature monitoring point #3 (southeast fuel box exit), MLP# 05-17 to address security system repairs and MLP #05-18 to address deep well pump and casing replacement with system upgrade. All four of these recent maintenance items were opened in July/August 2005 and are expected to be closed out relatively early in the next reporting year though the thermocouple/ temperature monitoring system failure may necessitate unstacking the shielding.

TABLE V-1

CHRONOLOGICAL TABULATION AND DESCRIPTION OF SCHEDULED UFTR SURVEILLANCES, CHECKS AND TESTS

Date		Surveillance/Check/Test Description
1 Sep 04	S-12	Review of Requalification Training Program Binders (Due 1 July 2004).
10 Sep 04	Q-6	Check of Posting Requirements (Due 31 August 2004).
15 Sep 04	Q-4	Radiological Survey of Unrestricted Areas (Due 30 September 2004).
15 Sep 04	Q-5	Radiological Survey of Restricted Areas (Due 30 September 2004).
16 Sep 04	Q-1	Check of Scram Functions (Due 30 September 2004).
17/30 Sep 04	S-4	Measurement of Argon-41 Stack Concentration (Includes Measurement of Dilution Air Flow Rate) (Due 26 August 2004).
28/30 Sep 04	A-1	Instrument and Test Equipment Calibration (Package and Send FLUKE Scopemeter to TRANSCAT for Calibration Check – Extra) (Not Due)
30 Sep 04	Q-7	Check of UFTR Building Fire Alarm System (Zone 3 – Upstairs Offices and Laboratories) (Due 23 September 2004).
1 Oct 04	Q-3	Radiological Emergency Evacuation Drill (Due 30 September 2004).
7 Oct 04	Q-10	Temperature Monitor/Recorder Data Transfer for Storage (Due 1 October 2004).
8 Oct 04	Q-8	Quarterly Report of Safeguards Events (Due 1 October 2004).
8 Oct 04	S-6	UFTR Semiannual Security Plan Key Inventory (Due 1 October 2004).
8 Oct04	S-3	Semiannual Inventory of Special Nuclear Material (Due 1 October 2004).
15 Oct 04	S-7	Semiannual Check (Replacement) of Security System Batteries (Due 7 October 2004).
19 Oct 04	A-1	Instrument and Test Equipment Calibration (Return of FLUKE Scopemeter from TRANSCAT following Calibration Check – Extra) (Not Due)
20 Oct 04	Q-2	Calibration Check of Area and Stack Radiation Monitors (Due 27 October 2004).
20 Oct 04	S-8	Leak Check of Neutron Sources (Due 31 October 2004).

TABLE V-1

CHRONOLOGICAL TABULATION AND DESCRIPTION OF SCHEDULED UFTR SURVEILLANCES, CHECKS AND TESTS

Date		Surveillance/Check/Test Description
22 Oct 04	Q-9	Calibration Check of Air Particulate Detector (AMS[4]) (Due 28 October 2004).
22 Oct 04	Q-9	Calibration Check of Air Particulate Detector (AIM3BL) (Due 28 October 2004).
31 Oct 04	A-5	Update of UFTR Decommissioning Cost Estimate (Due 31 July 2004).
29 Nov 04	Q-6	Check of Posting Requirements (Due 30 November 2004).
10 Dec 04	Q-5	Radiological Survey of Restricted Areas (Due 15 December 2004).
10 Dec 04	Q-4	Radiological Survey of Unrestricted Areas (Due 15 December 2004).
14 Dec 04	Q-7	Check of UFTR Building Fire Alarm System (Zone 4 – Annex Upstairs and Downstairs) (Due 30 December 2004).
16 Dec 04	S-9	Replacement of Deep Well Secondary Pump Fuses (Due 31 December 2004).
16 Dec 04	Q-1	Check of Scram Functions (Due 16 December 2004).
16 Dec 04	S-7	Semiannual Check (Replacement) of Security System Batteries (Partial to Replace 4 Volt Rechargeable Batteries) (Not Due).
17 Dec 04	Q-3	Radiological Emergency Evacuation Drill (Large Annual Drill Involving Outside Agencies) (Due 31 December 2004).
17 Dec 04	S-1	Measurement of Control Blade Drop Times (Due 31 December 2004).
17 Dec 04	S-5	Measurement of Control Blade Controlled Insertion Times (Due 31 December 2004).
17 Dec 04	S-11	Replacement of Control Blade Clutch Current Light Bulbs (Due 31 December 2004).
20 Dec 04	Q-6	Check of Posting Requirements (Partial to Post Updated Appointment Letter for RSRS Members and Alternates) (Not Due).
23–30 Dec 04	B-5	Evaluation and Recertification of Licensed Operators (Due 31 December 2004).
28 Dec 04	Q-6	Check of Posting Requirements (Partial to Post Updated Agreement Letter for Alachua County Division of Emergency Management) (Not Due).

TABLE V-1

CHRONOLOGICAL TABULATION AND DESCRIPTION OF SCHEDULED UFTR SURVEILLANCES, CHECKS AND TESTS

Date		Surveillance/Check/Test Description
2 Jan 05	Q-8	Quarterly Report of Safeguards Events (Due 1 January 2005).
14 Jan 05	S-10	Check and Update of Emergency Call Lists (Partial to Update Call Lists) (Due 31 December 2004).
14 Jan 05	Q-3	Radiological Emergency Evacuation Drill (Large Annual Drill Involving Outside Agencies) (Correction of Some Documents) (Due 31 December 2004).
23 Jan 05	Q-11	West Lot Integrity Checks
25 Jan 05	Q-2	Calibration Check of Area and Stack Radiation Monitors (Due 20 January 2005).
25 Jan 05	A-7	Visual Inspection of Emergency SCBA MSA Model 401 Tanks (Due 6 January 2005).
31 Jan 05	S-2	Annual Reactivity Measurements (Worth of Control Blades, Total Excess Reactivity, Reactivity Insertion Rate and Shutdown Margin) (Early, Not Due; Partial – Measurements Completed for Reg Blade)
1–22 Feb 05	S-2	Annual Reactivity Measurements (Worth of Control Blades, Total Excess Reactivity, Reactivity Insertion Rate and Shutdown Margin) (Completion) (Due 28 February 2005).
2 Feb 05	Q-9	Calibration Check of Air Particulate Detector (AIM3BL) (Due 22 January 2005).
7 Feb 05	Q-9	Calibration Check of Air Particulate Detector (AMS[4]) (Due 22 January 2005).
9 Feb 05	Q-10	Temperature Monitor/Recorder Data Transfer for Storage (Due 1 January 2005) (Overdue but Allowed by Indicated Storage Capacity)
18 Feb 05	S-10	Check and Update of Emergency Call Lists (Completion – Checking and Posting Updated Call List #2) (Due 31 December 2004).
21 Feb 05	S-7	Semiannual Check (Replacement) of Security System Batteries (Partial to Replace 4 Volt Rechargeable Batteries) (Not Due).
24 Feb 05	S-4	Measurement of Argon-41 Stack Concentration (Includes Measurement of Dilution Air Flow Rate) (Due 28 February 2005).

TABLE V-1

CHRONOLOGICAL TABULATION AND DESCRIPTION OF SCHEDULED UFTR SURVEILLANCES, CHECKS AND TESTS

Date		Surveillance/Check/Test Description
1 Mar 05	S-12	Review of Requalification Training Program Binders (Due 1 January 2005).
2–8 Mar 05	A-2	UFTR Nuclear Instrumentation Calibration Check and Calorimetric Heat Balance (Due 28 February 2005).
7 Mar 05	S-9	Replacement of Deep Well Secondary Pump Fuses (Extra Due to Trip) (Not Due).
10 Mar 05	Q-4	Radiological Survey of Unrestricted Areas (Due 10 March 2005).
10 Mar 05	Q-5	Radiological Survey of Restricted Areas (Due 10 March 2005).
10/14 Mar 05	Q-1	Check of Scram Functions (Due 16 March 2005).
14 Mar 05	Q-6	Check of Posting Requirements (Due 28 February 2005).
22 Mar 05	Q-3	Radiological Emergency Evacuation Drill (Due 17 March 2005).
24 Mar 05	A-4	Check/Replacement of Fire Alarm System Monitoring Station Batteries (Due 31 March 2005).
24 Mar 05	Q-7	Check of UFTR Building Fire Alarm System (Zone 1 – Reactor Cell and Control Room) (Due 17 March 2005).
7 Apr 05	S-3	Semiannual Inventory of Special Nuclear Material (1 April 2005).
11 Apr 05	Q-10	Temperature Monitor/Recorder Data Transfer for Storage (Due 1 April 2005).
12 Apr 05	Q-8	Quarterly Report of Safeguards Events (Due 1 April 2005).
14 Apr 05	S-6	UFTR Semiannual Security Plan Key Inventory (1 April 2005).
14 Apr 05	A-6	Physical Inventory for Security-Related Locks/Cores (Due 31 March 2005).
15 Apr 05	S-7	Semiannual Check (Replacement) of Security System Batteries (Partial to Replace Certain Alarm Batteries) (Not Due).
22 Apr 05	Q-6	Check of Posting Requirements (Partial to Post and Assure Posting of RCO Notification Posters) (Not Due).

TABLE V-1

CHRONOLOGICAL TABULATION AND DESCRIPTION OF SCHEDULED UFTR SURVEILLANCES, CHECKS AND TESTS

Date		Surveillance/Check/Test Description
25 Apr 05	Q-11	West Lot Integrity Checks (23 April 2005).
26 Apr 05	B-3	Review of UFTR Standard Operating Procedures Manuals for Completeness (Completion of Review Information Copy #4 for RCO and Work on Information Copy #1 for NRE Chairman) (Due 30 September 2003 – Except Information Copies).
12 May 05	S-8	Leak Check of Neutron Sources (20 April 2005).
12 May 05	Q-2	Calibration Check of Area and Stack Radiation Monitors (Due 25 April 2005).
12 May 05	Q-9	Calibration Check of Air Particulate Detector (AIM3BL) (Due 30 April 2005).
16 May 05	Q-9	Calibration Check of Air Particulate Detector (AMS[4]) (Due 30 April 2005).
17 May 05	A-1	Instrument and Test Equipment Calibration (Shipment of FLUKE Series 3 Voltmeter, OMEGA TEMP Model HH23 and Kurz Minianemometer 491) (Due 31 March 2005).
31 May 05	A-3	Annual Measurement of UFTR Temperature Coefficient of Reactivity (Due 31 March 2005).
3 Jun 05	A-1	Instrument and Test Equipment Calibration (Receipt of FLUKE Series 3 Voltmeter and OMEGA TEMP Model HH23) (Due 31 March 2005).
14 Jun 05	Q-4	Radiological Survey of Unrestricted Areas (Due 10 June 2005).
18–25 Jun 05	B-6	Evaluation of Emergency Plan (Due 26 December 2004).
20–21 Jun 05	S-7	Semiannual Check (Replacement) of Security System Batteries (Delayed but Functional Test Verified Operability) (15 April 2005).
27 Jun 05	Q-5	Radiological Survey of Restricted Areas (Due 10 June 2005).
29 Jun 05	Q-6	Check of Posting Requirements (Due 31 May 2005).

TABLE V-1

CHRONOLOGICAL TABULATION AND DESCRIPTION OF SCHEDULED UFTR SURVEILLANCES, CHECKS AND TESTS

Date		Surveillance/Check/Test Description
1 Jul 04	Q-1	Check of Scram Functions (Due 25 June 2004).
2 Jul 04	Q-4	Radiological Survey of Unrestricted Areas (Due 5 June 2004).
2 Jul 04	Q-5	Radiological Survey of Restricted Areas (Due 22 June 2004).
7 Jul 05	S-9	Replacement of Deep Well Secondary Pump Fuses (Due 16 June 2005).
8/30 Jul 05	S-10	Check and Update of Emergency Call Lists (Due 30 June 2005).
11/22 Jul 05	Q-1	Check of Scram Functions (Due 10 June 2005).
14 Jul 05	Q-7	Check of UFTR Building Fire Alarm System (Zone 2 – Downstairs Offices and Labs) (Due 24 June 2005).
9/23 Jul 05	A-1	Instrument and Test Equipment Calibration (Receipt of Kurz Minianemometer Unchecked with Reshipment to Alternate Calibration Company) (Due 31 March 2005).
20 Jul 05	Q-3	Radiological Emergency Evacuation Drill (Due 22 June 2005).
21 Jul 05	Q-6	Check of Posting Requirements (Partial to Post Memorandum Updating Those Authorized to Carry Cell Keys for Drills and Emergencies) (Not Due).
29 Jul 05	Q-8	Quarterly Report of Safeguards Events (Due 1 July 2005).
29 Jul 05	Q-10	Temperature Monitor/Recorder Data Transfer for Storage (Due 1 July 2005).
3 Aug 05	Q-2	Calibration Check of Area and Stack Radiation Monitors (Due 31 July 2005).
3 Aug 05	Q-9	Calibration Check of Air Particulate Detector (AIM3BL) (Due 31 July 2005).
15 Aug 05	Q-11	West Lot Integrity Checks (Due 25 July 2005).
26 Aug 05	Q-9	Calibration Check of Air Particulate Detector (AMS[4]) (Due 31 July 2005).
31 Aug 05	A-1	Instrument and Test Equipment Calibration (Receipt of Kurz Minianemometer 490) (Due 31 March 2005).

Note: An asterisk is used to indicate the surveillance was not completed within the allowable interval resulting in reactor unavailability for normal operations.

TABLE V-1

CHRONOLOGICAL TABULATION AND DESCRIPTION OF SCHEDULED UFTR SURVEILLANCES, CHECKS AND TESTS

Date	Surveillance/Check/Test Description

Note: Required UFTR surveillances, checks and tests are up to date at the end of the reporting year. In some years, surveillances have been carried over to the new year within the allowable interval; such was the case last year for the Q-1, Q-3, Q-4, Q-5, Q-6, Q-7, S-4, S-12, A-5, V-1 and V-2 surveillances, all of which were subsequently completed within the required interval or with the case of V-1 and V-2 surveillances, the UFTR Tech Specs were changed to extend the interval so they became X-1 and X-2 surveillances. The surveillances carried over this year include the Q-1, Q-3, Q-4, Q-5, Q-6, Q-7, S-1, S-4, S-5, S-11, S-12, A-5, B-1, B-3 and B-4 surveillances with several not completed within the required interval due to the continuing outage to address the high PC temperature trip and related maintenance.

TABLE V-2

CHRONOLOGICAL TABULATION OF UFTR PREVENTIVE/CORRECTIVE MAINTENANCE

Date	Maintenance Description
16 Mar 94	After the new area radiation monitoring system including a 19-inch rack, recorder, computer console, battery backup, probes, attachments, cabling and hardware was received, MLP #94-14 was used to control setup of the new ARM system including connecting the battery power supply and the recording module. During April 1994, the new detectors were also mounted. During May 1994, electrical cables were run from the detectors to the control room monitors. Actual on-line installation of the new system will require a modification package which is partially prepared. No work has been accomplished since May 1994, again primarily because of relatively trouble-free operation though the recent problems with the North ARM under MLP #03-06 and the East ARM under MLP #03-25 may indicate a need to reconsider as some time was spent in June 2003 performing bench top checks of the new B-91-9111 Microdata Logger system. (MLP #94-14 remains open.)
11 Nov 96	Following one spurious security alarm on November 10 and two alarms on November 11, 1996, the security system batteries were checked and replaced (S-7 Surveillance). Under MLP #96-30 the rechargeable batteries were found to be low and were recharged. Subsequently, 10 CFR 50.59 Evaluation Number 96-13 was developed to allow modification and replacement of the power pack to prevent recurrence of the problem of spurious alarms due to low voltage. Measurements were made and security system circuits checked and verified. In addition, the 6-volt batteries were recharged in mid-month. At the end of November 1996, design and development of a new power pack per 10 CFR 50.59 Evaluation Number 96-13 were in progress. At the end of December 1996, the 10 CFR 50.59 Evaluation is complete as is the design. On January 7, 1997, the new power supply was installed with all but one siren operational to meet requirements. Subsequently, the west lot siren was repaired on January 13 and both the west lot and journalism side siren horn driver wiring was reterminated on January 14, 1997. Drawings and maintenance log were subsequently updated and an evaluation made that separate grounds would be needed for the security system batteries to assure proper charging and eliminate spurious alarms as the batteries discharge over time. On March 10, 1997, the power supply was removed for modification. Upon installation, various problems occurred resulting in partial and intermittent compensated outage of the security system over the period March 1021 with circuit mapping performed for troubleshooting on March 19 and the intermittent ground finally repaired on

Date	Maintenance Description
	March 21, 1997, but without installation of the modification to separate grounds, basically returning the system to its state prior to March 10. Subsequently, the 4-volt rechargeable batteries have been replaced on May 14, June 18, July 7, and July 24, 1997 (for prevention purposes on July 30, 1997), on August 29, and on September 29, 1997. Following a full S-7 Surveillance on October 24, 1997, the loss of the holdup alarm was corrected under MLP #96-30 by reterminating a loose wire. Subsequently, the 4-volt rechargeable batteries were replaced on December 16, 1997. The 4-volt rechargeable batteries were replaced again on January 9, February 10, March 10, April 8, and on May 6, 1998. Following a full S-7 Surveillance on May 27, 1998, the 4-volt rechargeable batteries were replaced again on June 24, July 24, August 19, September 16 and October 13, 1998. Following a full S-7 Surveillance including replacement of rechargeable batteries on November 10, the 4-volt rechargeable batteries were replaced again on December 7, 1998. The 4-volt rechargeable batteries were replaced again on January 4, February 1 and March 2, 1999 with upgraded 4-volt batteries installed on March 12, 1999 under 10 CFR 50.59 Evaluation Number 99-02 developed and approved in February to upgrade the 4-volt rechargeable batteries for longer life. There had been no need for further replacement through the end of July 1999 though the full S-7 Surveillance was performed on July 2, 1999. Following the full S-7 Surveillance, when the 4-volt batteries were not replaced, the 4-volt rechargeable batteries were replaced again on August 24 and November 5, 1999, while a full S-7 Surveillance without replacement of the 4-volt batteries was conducted on November 11, 1999. The 4-volt rechargeable batteries were replaced again on February 24, 2000. There had been no further need for replacement until completion of the full S-7 Surveillance on May 25, 2000. The 4-volt rechargeable batteries were again replaced on November 10, 2000 followed by a full S-7 Surveillance on December 29, 2000. The 4-volt rechargeable batteries were replaced again on February 26, 2001. There had been no further need for replacement until completion of the full S-7 Surveillance on May 22, 2001. Subsequently the 4-volt rechargeable batteries were replaced again on August 24 and on November 9, 2001 while a full S-7

TABLE V-2

CHRONOLOGICAL TABULATION OF UFTR PREVENTIVE/CORRECTIVE MAINTENANCE

Date	Maintenance Description
	Surveillance without replacement of the 4-volt batteries was conducted on December 3, 2001. Subsequently, the 4-volt rechargeable batteries were replaced again on January 16 and on March 29, 2002 while a full S-7 Surveillance was conducted on June 6, 2002. Subsequently, the holdup alarms' batteries were replaced due to low voltage on August 16, 2002 and the 4-volt rechargeable batteries were replaced again on August 21, 2002. Current plans are to replace the entire system with an equivalent one with DOE 20012 URI grant funds. A full S-7 Surveillance was conducted on October 28/31, 2002. Subsequently, the 4-volt rechargeable batteries were replaced again on January 2 and on March 11, 2003, with another full S-7 Surveillance conducted on April 25, 2003. There had been no further need for replacement until completion of the full S-7 Surveillance on November 7, 2003. Subsequently, the 4-volt rechargeable batteries were replaced again on December 11, 2003. The 4-volt rechargeable batteries were replaced again on January 22, 2004 with another full S-7 Surveillance conducted on April 7, 2004. Subsequently, the 4-volt rechargeable batteries were replaced again on May 6, 2004 and on July 14, 2004 with another full S-7 Surveillance conducted on October 15, 2004 without replacement of the 4-volt rechargeable batteries. Subsequently, the 4-volt rechargeable batteries were replaced again on December 16, 2004. The 4-volt rechargeable batteries were replaced again on February 21 and April 15, 2005 with another full S-7 Surveillance conducted on June 2021, 2005. (MLP #96 30 remains open.)
14 Oct 02	During some counting checks, the single channel analyzer (SCA) and timer/counter modules on the counting experiment equipment rack were noted to be giving spurious counting results. Under MLP #02-26, the portable bin SCA and timer/counter modules were transferred to the NRE electronics engineer under an NRE work request for troubleshooting and repair with no results to date as the modules were returned with no work performed before the electronics engineer's last workday on January 31, 2003. (MLP #02-26 remains open.)
5 Aug 04	For some time, plans had been made to conduct work in the west lot for the Weil Replacement Chill Water project including meetings, discussions and visits for measurements by and for various personnel including UF PPD Project Manager Jeff Bair, PPI supervisor Troy Lauramore, Matt Searles of Perry Construction and

Date	Maintenance Description
	foreman Jimbo Williams of WW Gay. Initial work began under MWO #0674958 on August 5, 2004 with temporary movement of the north section of the west lot fence to allow clearing room for pipes under supervision of WW Gay foreman Jimbo Williams. No further work was accomplished inside the fenced area until August 18, 2004. Subsequently, holes were drilled in the reactor cell west wall under 10 CFR 50.59 Evaluation and Determination Number 04-01 (Modification/Upgrade of Chilled Water System for Reactor Building Pipe Supports/Wall Anchors) to provide anchoring for the chill water line pipe supports on August 19, 2004. The minutes of the August 19, 2004 meeting of the RSRS Executive Committee which reviewed and approved these anchors is Attachment II to the August 2004 report. With installation of the main chill water pipes below ground leading into the west lot, the west lot fence was restored to its original location in improved condition on August 20, 2004 as Tom Quarles of Florida Enterprise Corporation replaced the barbed wire as well. At the end of August, the main piping was in place and efforts involving daily access were under way to complete the pipe installation. During September 2004, the chilled water line replacement project continued with the usual vehicular, equipment, supplies and personnel access until September 14 when most work was completed and personnel were removed to another project. Perry Construction supervisor Matt Searles accessed the west lot on September 15, 16 and 22 to perform some grout work with no further work accomplished through the end of September. On October 11, 2004, Troy Lauramore and then Matt Searles contacted the facility concerning access for some work on October 21-22 prior to hook up of the new chill water lines scheduled for October 23, 2004. Matt Searles then arranged for access and Jimbo Williams of WW Gay then accessed the west lot with Richard Ellison and Matthew Kight to make final measurements. On October 19 plans for final hookup on October 23 were verified with Matt Searles and discussions were conducted with PPD alarm systems supervisor Skip Rockwell concerning fire alarm cut off requirements. Subsequently, PPD alarms system technician Tom Boynton visited and verified that the chill water final hookup would not require bypassing any portions of the fire alarm system. Subsequently, university architect Chandler Rozier and COE Services supervisor Denis Mercier walked through to check on the west lot chiller work and plans for the reactor building annex remodeling. As planned, on October 23, PPD supervisor Mike Vaughn, Perry Construction supervisor Matt Searles and WW Gay Foreman Jimbo Williams and various support personnel (William Brown, Matthew Kight, Caleb Smith, Lance Hersey and forklift

Date	Maintenance Description
	operator Don Heinz) accessed the west lot with various equipment and performed final hookup installation of the new chill water lines. Subsequently, on October 24, Matt Searles made arrangements for final insulation installation with Gaylord Insulation personnel (Steve Taylor, Arthur Gilbert and Paul Videon) accessing the west lot on October 26 and 27 to perform the work. At the end of October little remained to be accomplished on this project with a substantial completion inspection planned for November. On November 1, 2004, Michael Ellison of WWGay accessed the west lot twice to perform some cosmetic repairs. Finally, on November 10, the substantial completion inspection was accomplished with those present including Matt Webster and Matt Searles of Perry Construction, Jimbo Williams of WW Gay, and Supervisor Mike Vaughn, Senior Engineer Gray Rawls, Project Manager Jeff Bair, and Charles Milford of PPD. (On 10 November 2004, MWO #0674958 was closed).
25 Aug 04	For several weeks the facility gas flow proportional counter system (PCC 11T/7508) was noted to be out of order due to a failed voltage adjustment potentiometer. Under MLP #04-18, the failed voltage adjustment potentiometer was replaced with an on-hand spare removed from another nonworking system. At the end of August 2004, the facility gas flow proportional counter system appeared to be operable but was awaiting proper calibration prior to use. Subsequently, during a precalibration check, the voltage adjustment potentiometer was found to be the wrong replacement and an electronics engineer was contracted through the Radiation Control Office to effect repairs. Repair work was awaiting receipt of spare parts at the end of September. Subsequently, on October 6, 2004 the voltage adjustment potentiometer was replaced by the electronics engineer with the necessary calibration checks conducted successfully by RCT J. Parker on October 8. The gas flow proportional counter system was then returned to service with no further problems noted. (On 8 October 2004, MLP #04-18 was closed.)
25 Aug 04	After noting some small decrease in the stack dilute fan rpm indication, MWO #0812197 was opened with two PPD mechanical technicians (Ross Henderson and Jesse Fleming) visiting on August 26 and returning on August 27 to replace filters on stack dilute fan intake room and to identify and order the proper replacement belts for the dilute fan drive motor. At the end of August, PPD was awaiting delivery of the replacement belts. Subsequently, MLP #04-20 was opened and replacement fan belts were installed and the bearings greased on

TABLE V-2

CHRONOLOGICAL TABULATION OF UFTR PREVENTIVE/CORRECTIVE MAINTENANCE

Date	Maintenance Description
	September 24 with a slight gain in rpm indication but much smoother operation with no further problems noted. (On 24 September 2004, MLP #04-20/MWO #0812197 were closed.)
1 Sep 04	After noting higher than normal upper level reactor cell room temperatures, Physical Plant Division was requested to change air filters and perform preventive maintenance checks on the cell air handler per MWO #0813185. Subsequently, PPD mechanical technician Jesse Fleming visited on September 9 to schedule the work. On September 10, Mr. Fleming replaced the filters and performed the necessary preventive maintenance checks with no problems noted though cell room temperature was somewhat reduced later with no problems noted. (On 10 September 2004, MWO #0813185 was closed.)
13 Sep 04	Shortly after lunch on September 13, following a violent noontime thunderstorm, several gallons of water were noted on the reactor cell floor from leaks along the south cell wall. The water was easily mopped up. Under MWO #0814198, PPD technician Dennis Ingram responded to check the situation in the cell but, fearing a storm, did not access the roof. Subsequently, the Director and RCO accessed the roof and cleared a clog in one of the drains on the cell roof. After a call to PPD's Willie Hill, PPD technician Bruce Sims responded later in the afternoon on September 13 and, after checking the cell roof from inside, accessed and checked the roof noting it appeared to need work. Subsequently, PPD technician Emerson Perry checked the roof from the cell on September 14 and then spent the day repairing copper flashing on the office roof section in the morning and along the leaking south cell roof during the afternoon. He then returned on September 15 and assured the flashing work was complete and that all roof drains were clear with no further problems noted. (On 15 September 2004, MWO #0814198 was closed.)
1 Oct 04	In checking the RM-14/2907 frisker unit, it was noted to be responding erratically and alarming periodically. There had been several previous alarms but the meter had checked satisfactorily. Under MLP #04-21, the GM tube in the frisker detector was replaced with an on-hand spare and checked to be responding properly. The frisker was then transferred to radiation control for a calibration check. Following a satisfactory calibration check, the frisker was returned to the reactor facility and returned to service with no further problems noted. (On 5 October 2004 MLP #04-21 was closed.)

TABLE V-2

CHRONOLOGICAL TABULATION OF UFTR PREVENTIVE/CORRECTIVE MAINTENANCE

Date	Maintenance Description
5 Oct 04	In September 2004, a door buzzer had alarmed inadvertently on one occasion and had been reset after verifying there was no valid reason for the alarm. This buzzer alarmed again on October 5, 2004 prior to removing the cell from security. Again it was reset after verifying there was no valid reason for the alarm. Subsequently, under MLP #04-22, an adhesive bumper was installed to prevent inadvertent buzzer actuation with no further problems noted. (On 5 October 2004, MLP #04-22 was closed.)
6 Oct 04	For several days there had been a tapping noise heard periodically throughout the upstairs area of the reactor building thought possibly due to a failing air handler compressor. When the source could not be located, MWO #0818607 was called in with a PPD mechanical technician looking for the source on October 6, 2004 and accessing the roof for checks on October 8, 2004. Subsequently, on October 14, with assistance from building occupants, the noise was isolated to the exhaust fan for the downstairs classroom area fume hood which runs periodically which made isolation of the noise source difficult. A temporary fix was implemented to correct the problem with the PPD mechanical technician noting new bearings will be installed in the future but no further problems noted. (On 14 October 2004, MWO #0818607 was closed.
6 Oct 04	During a periodic check, the E140/911 survey meter was noted to be responding spuriously whenever scales were changed. Under MLP #04-23, the meter was checked and some contacts cleaned on October 6 but the problem persisted. On October 7, 2004, the meter was disassembled and electric contract spray utilized to clean contacts but still the problem persisted so the meter was transferred to the radiation control technician for repair. Subsequently, various contacts were cleaned two more times finally correcting the spurious response. Following successful calibration checks on October 21, the survey meter was returned to the facility and returned to normal service with no further problems noted. (On 21 October 2004, MLP #04-23 was closed.)
19 Oct 04	For some time the east area radiation monitor chart recorder would be noted to stop moving periodically with the function usually able to be restored by assuring proper gear engagement. Under MLP #04-24, the gearbox for the east area radiation monitor recorder was replaced on October 19, 2004 with an on-hand spare to restore reliable proper recording function with no further problems noted. (On 19 October 2004, MLP #04-24 was closed.)

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Date	Maintenance Description
25 Oct 04	Following the hookup of the new building chill water system replacement on October 23, 2004, the entire reactor building was noted to be excessively cold, essentially at chilled water temperatures. Simultaneously, the Nuclear Science Building was noted to be excessively hot. Under MWO #0821416, various PPD personnel including an electrician, a supervisor, two alarm systems technicians as well as three mechanical technicians finally corrected a misaligned valve in the main equipment room to restore proper temperature control for the Nuclear Science Building and a PPD controls technician replaced a failed temperature control in the reactor building equipment room to restore proper temperature control for the reactor building by late afternoon on October 25, 2004 with no further problems noted. (On 25 October 2004, MWO #0821416 was closed.)
3 Nov 04	During the weekly checkout, the shield tank water was noted to be nearing the lower limit allowed on resistivity. Under MLP #04-25 both shield tank water recirculation system filter/demineralizer cartridges were replaced with spares to assure high resistivity of shield tank water with no further problems noted. (On 3 November 2004, MLP #04-25 was closed.)
9 Nov 04	During the weekly checkout on November 9, 2004, the primary coolant storage tank level was noted to be nearing replacement level. Under MLP #04-26, 36 gallons of demineralized water were added to the tank to raise the level from 20¾ inches to 26 inches to restore proper level with no problems noted. (On 9 November 2004, MLP #04-26 was closed.)
29 Nov 04	In the evening on November 28 and again on the morning of November 29, 2004, a buzzing sound (not a trouble alarm) was noted to be emanating from the fire alarm monitoring station outside Room 108 NSC. Under MWO #0826888, PPD personnel visited and corrected the noise with no further problems noted. (On 29 November 2004, MWO #0826888 was closed.)
10 Dec 04	In late morning on December 10, 2004, the temperature and humidity in the facility were noted to be higher than usual. A maintenance work order (MWO #0829816) was called in and the facility was informed some checks were in progress in anticipation of completing the Weil Hall chiller work in the December 20, 2004 to January 3, 2005 holiday period. In early afternoon conditions returned to normal. Subsequently, a PPD technician visited on

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Date	Maintenance Description
	December 13 and was told conditions were satisfactory. (On 13 December 2004, MWO #0829816 was closed.)
15 Dec 04	During usage of the NAA Lab door to the west lot, the cell lock deadbolt was discovered to be unresponsive. A maintenance work order (MWO #0831481) was called in and a PPD locksmith responded and evaluated the problem as the bolt rubbing against the housing which could be corrected by reactor staff. Under MLP #04-27, the housing was filed down and the deadbolt lock verified to be working satisfactorily with no further problems noted. (On 15 December 2004, MLP #04-27/MWO #0831481 were closed.)
16 Dec 04	During a cell walk-through on December 14, 2004, the demineralized city water line was discovered to have been left running so the resistivity level of the makeup water provided by the demineralizer system was checked and noted to indicate end of life for resins. Under MLP #04-28, the resins were replaced to restore the source of high resistivity makeup water for the reactor facility with no problems noted. (On 16 December 2004, MLP #04-28 was closed.)
21 Dec 04	When the deep well pump tripped out due to a blown fuse, it was decided to try operating with the city water cooling system. However, several checks of the city water trip were unsuccessful due to the flow meter valve not closing completely. Subsequently, under MLP #04-29, the actuator on the flow meter valve was lubricated but there was little effect on the operation with no further work attempted by the end of December 2004. In April 2005, a replacement flow meter was identified to avoid failure problems; this meter was ordered on May 2 and finally received on May 9, 2005. Subsequently, plans were made to install the new flow meter with 10 CFR 50.59 Evaluation Number 05-02 (Upgrade Replacement of UFTR City Water Flow Meter) developed to control the installation. All the necessary support materials were assembled and installation of the new flow meter was undertaken beginning on May 24 and continuing to May 25 to correct initial minor leaks and assure proper operation with subsequent completion of documentation with several subsequent verifications of proper flow meter operation with no further problems noted. (On 25 May 2005, MLP #04-29 was closed.)

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Date	Maintenance Description
23 Dec 04	During performance of the daily checkout, the secondary cooling system flow meter indicator was noted to be sticking. Under MLP #04-30, the meter internals were cleaned to restore proper operation as verified by several checks with no further problems noted. (On 23 December 2004, MLP #04-30 was closed.)
29 Dec 04	During performance of the daily preoperational check, the senior reactor operator noted the AMS4 air particulate detector chart recorder was slipping. Under MLP #04-31, a worn gear on the recorder drive was replaced with an on-hand spare and proper recorder movement verified with no further problems noted. (On 29 December 2004, MLP #04-31 was closed.)
5 Jan 05	In mid afternoon on January 5, 2005, the temperature in the reactor cell and NAA Lab were noted to be much higher than usual. A maintenance work order (MWO #0833547) was called in. When the problem remained by mid afternoon on January 6, it was called in again. Subsequently, three PPD technicians arrived and verified proper operation of the ex-cell air handler. The latter two technicians then verified the cell air handler as well. Subsequently, the technicians corrected a misaligned valve outside the building to restore proper cooling by late afternoon with no further problems noted. (On 6 January 2005, MWO #0833547 was closed.)
11 Jan 05	For several weeks the rpm indication on the stack fan had been reducing slightly. On January 11, 2005, MWO #0834777 was called in. Subsequently, a PPD mechanical technician visited on three occasions but failed to schedule or was delayed, so it was not until January 24 that he arrived and, with the fan secured, repositioned the motor and tightened the belts to improve the rpm indication. The PPD mechanical technician also noted the motor was drawing higher amperage than recommended (14.8 amps versus 13.2 + 6% amps) so he was to check with his supervisor and get back with a recommendation. In the interim, the system continued to operate satisfactorily. (On 24 January 2005, MWO #0834777 was closed.)
8 Feb 05	During the weekly checkout on February 8, 2005, the primary coolant storage tank level was noted to be nearing replacement level. Under MLP #05-01, 42 gallons of demineralized water were added to the tank to raise the level from 20¼" to 26¾" to restore proper level with no problems noted. (On 8 February 2005 MLP #05-01 was closed.)

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Date	Maintenance Description
8 Feb 05	For several days the lock on the door to the staff offices was not functioning so a work order was called in to PPD. Under MLP #05-02/MWO #0840279, PPD locksmith Jim Beck repaired and refurbished the lock on February 8, 2005 with closeout of MLP #05-02/MWO #0840279. Subsequently the lock failed again on February 18, so another work order was called in. Under reopened MLP #05-02 and MWO #0842203, a PPD locksmith and an assistant removed the failed lock and replaced it with a new lock with no further problems noted. (On 18 February 2005, MLP #05-02/MWO #0840279 and MWO #0842203 were closed.)
2 Mar 05	During performance of the pre-calorimetric portion of the A-2 Surveillance (UFTR Nuclear Instrumentation Calibration Check and Calorimetric Heat Balance), certain voltages and setpoints were noted to require minor adjustments as expected. Under MLP #05-03, various voltages and setpoints were adjusted to assure proper nuclear instrumentation calibration with no further changes needed at the conclusion of the post-calorimetric checks. On March 3, 2005, following performance of various initial adjustments and checks as part of the annual UFTR Nuclear Instrumentation Calibration Check and Calorimetric Heat Balance (A-2 Surveillance), a reactor startup was commenced at 1201 for the first power run to allow generation of sufficient neutron level per SOP-E.4, Step 7.1.9. At 1223 hours, with reactor power at 30 kW, the operator noted primary coolant inlet temperature reading 77.4 °F and primary outlet temperature reading 74.2 °F indicating the two points had apparently been reversed during the prior adjustments and checks. An unscheduled shutdown was commenced at 1223 hours with the reactor shut down and secured at 1225 hours and all systems responding properly. Subsequently, under MLP #05-03 applicable for the A-2 Surveillance adjustments and checks, the thermocouple connections were checked; Point #6 (NE box) and Point #7 (inlet) thermocouple connections were found to have been switched. The connections were then placed in the correct positions and all connections verified (Points #1#10) by the SRO and independently by an SRO-trainee. Following a successful daily checkout, UFTR Form SOP-0.6B (Unscheduled Reactor Shutdown Review and Evaluation) was completed and restart to continue the A-2 Surveillance (Step 7.1.9) was approved. It was noted that the switched thermocouple connections were realized quickly by an alert operator as soon as possible based on temperature differences; the misconnected

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Date	Maintenance Description
	switches had no impact on safety since all temperature points provide the same trip impact so there was negligible impact on safety due to the error. Subsequently, a second daily checkout was completed on March 4, 2005. During the subsequent restart, the backup Honeywell temperature monitor was noted to be producing erroneous readings but it was reinitialized to restore proper indication. The 6-hour run at full power was successful with shutdown commenced at 1527 hours on March 4. The reactor was shut down and secured with no problems noted at 1529 hours. (On March 8, 2005, MLP #05-03 was closed.)
7 Mar 05	On March 7, 2005, reactor startup was begun at 0911 hours to support the annual UFTR Nuclear Instrumentation Calibration Check and Calorimetric Heat Balance (A-2 Surveillance) with 100 kW full power reached at 0931 hours. Subsequently, a blade drop trip occurred at 1107 hours while the reactor was critical at 100 kW and running for approximately 1.5 hours in automatic control. The SRO operator noted the "SEC PRESS" indicator lit on the scram annunciator after the "Well Warning" and "Flow Scram" indicators energized, indicating loss of or low secondary water flow. In accordance with Technical Specifications Section 2.2, approximately 10 seconds later the "SEC PRESS" trip came in and tripped the reactor. At the same time the bottom indicator of the secondary well pump on/off switch energized, indicating a loss of power at that point. MLP #05-04 was then opened and the well pump fuses were immediately checked for continuity. The southernmost 60-amp well pump fuse was found to be blown and hot to the touch. The other fuses were working properly (continuity check 0.2 ohms). All three fuses were replaced and documented in the semiannual replacement of deep well secondary pump fuses (S-9 Surveillance). The overload reset button was also reset. The secondary well pump was then energized and left running for approximately 1.5 hours. No abnormalities were observed during this time and the well pump was secured. UFTR Form 0.6A (Unscheduled Reactor Trip Review and Evaluation) was also completed to approve restart. The subsequent restart and operation for over 7 hours on March 8 to complete the A-2 Surveillance was also successful. Based on the indications at the time of the blade drop trip, the well pump 60 amp fuse failed. The fuse failure resulted in a reduction in water flow and an increase

Date	Maintenance Description
	in amperage in the other two phases of electrical power. The reduction in water flow caused the "SEC PRESS" trip after a 10-second delay. The increase in amperage of the other two phases caused the overload circuit to trip and remove power to the secondary well pump.
	All safety and control systems were noted to have operated correctly and in accordance with the Tech Specs for the trip from a known cause. Based on this fact, this blade drop trip is evaluated as not promptly reportable as defined in Technical Specifications Section 6.6.2. This event is also evaluated as having negligible impact on reactor safety or on the health and safety of the public or reactor facility staff.
	The completed UFTR Form 0.6A (Unscheduled Reactor Trip Review and Evaluation) is Attachment I to the March 2005 monthly report. A memorandum describing the reactor trip event is Attachment II to the March 2005 monthly report. (On March 7, 2005, MLP #05-04 was closed.)
11 Mar 05	On March 10, 2005, during performance of the quarterly Check of Scram Functions (Q-1 Surveillance), the trip on loss of primary coolant level (item 4 on Page 2 of 6 of the Q-1 Surveillance Data Sheets) was unable to be completed by verifying a change in voltage reading. It was noted that this was a failure of the test mechanism and not a failure of the trip itself. After initial non-invasive investigation of the system, further investigation was made under MLP #05-05 and the decision was made to utilize a test lamp versus a voltage reading and to require that the scram level occur at ≥43.0" versus the previous 44.5" since it could be observed much more accurately with the test lamp installed. A modification package under 10 CFR 50.59 Evaluation and Determination Number 05-01 (Change in Method of Checking PC Level Trip on Quarterly Scram Checks) was prepared, reviewed and approved at a meeting of the RSRS Executive Committee on March 14, 2005 where it was noted that this change is for the method of verifying the scram check for the PC level; there is no change in the scram itself. Essentially, it was agreed that the method change represents an improvement in the method of performing the PC level scram check.
	Subsequently, after the approval at the RSRS Executive Committee meeting, the procedure change for the Q-1 Surveillance Data Sheets (Page 2 of 6, item 4, lines 4–6) was implemented, the changed test method (test lamp) was installed and the

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Date	Maintenance Description
	PC level trip was verified as required to complete the Q-1 scram checks with no further problems noted. This failure was noted to be for the test method only and to have had no impact on reactor safety or on the health and safety of the public or the reactor staff. The RSRS Executive Committee meeting minutes without the attachments but including the approval sheet and updated changed Page 2 of the Q-1 Surveillance Data Sheets constitute Attachment IV to the March 2005 monthly report. (On March 14, 2005, MLP #05-05 was closed.)
30 Mar 05	During performance of the daily preoperational check with observation by the NRC inspection team, the linear (red) pen of the two-pen recorder was noted to be unresponsive. The daily check was suspended and initial checks were uninformative. Under MLP #05-06, the source of the problem was isolated to the recorder itself whereupon a switch for input to the red pen channel was noted to be inadvertently in the "OFF" condition. When returned to the "ON" position, the linear (red) pen again returned to normal operation. At this point the reactor was essentially returned to normal operations though a successful daily checkout would still be needed to allow normal operations. This daily checkout was delayed due to NRC inspection and other commitments but completed successfully as expected on April 3, 2005 with no further problems noted. (On March 30, 2005, MLP #05-06 was closed.)
4 Apr 05	During the weekly checkout on April 3, 2005, the primary coolant storage tank level was noted to be nearing replacement level. Under MLP #05-07, 38 gallons of demineralized water were added to the tank to raise the level from 20¾" to 26½" to restore proper level with no problems noted. (On 4 April 2005, MLP #05-07 was closed.)
20 Apr 05	On April 19, 2005 in late afternoon, a minor problem occurred with part of the security system. The problem did not seem to be with the installation at the reactor facility. Subsequently, under MLP #05-08, the problem was verified on April 20 to be external to the installation at the reactor facility. The problem was verified to be corrected at 1535 hours on April 20 with no further problems noted. (On 20 April 2005, MLP #05-08 was closed.)

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Date	Maintenance Description
2 May 05	During performance of the daily checkout, the secondary cooling system flow meter indicator was noted to be sticking. Under MLP #05-09, the meter internals were cleaned to restore proper operation as verified by several checks with no further problems noted. (On 2 May 2005, MLP #05-09 was closed.)
6 May 05	For several days the lock in the upstairs door leading downstairs to the limited access area had been difficult to open. On May 6 in early afternoon the lock ceased to operate and MWO #0856606 was called in to PPD. A PPD locksmith arrived soon thereafter and verified the need for a new cylinder. Later in the afternoon a PPD locksmith returned with an assistant and installed a new cylinder in the lock to restore proper operation with no further problems noted. (On 6 May 2005, MWO #0856606 was closed.)
27 May 05	In the afternoon of May 27, 2005, the lock in the downstairs Nuclear Reactor Building door allowing access from the Nuclear Sciences Building came apart upon being operated and ceased working. Under MWO #0860669, a PPD locksmith reassembled and repaired the lock to restore proper operation with no further problems noted. (On 27 May 2005, MWO #0860669 was closed.)
14 Jun 05	During the weekly checkout on June 14, 2005, the resistivity level of the makeup water provided by the demineralizer system for the city water was noted to be indicating near end of life for the resins. Under MLP #05-10, the resins were replaced to restore the source of high resistivity makeup water for the reactor facility with no problems noted. (On 14 June 2005, MLP #05-10 was closed.)
14 Jun 05	During a weekly checkout in mid-June, the primary coolant storage tank level was noted to be nearing replacement level. Under MLP #05-11, 51 gallons of demineralized water were added to the tank to raise the level from 20¾" to 27¾" to restore proper level with no problems noted. (On 14 June 2005, MLP #05-11 was closed.)
16 Jun 05	On June 13, 2005, an effort was made to pump wastewater from the indoor storage tanks to the outside aboveground wastewater holdup tank. The transfer pump was out of commission due to a seized head and a replacement pump was ordered. However, under MLP #05-12, on June 16, 2005 the existing transfer pump head was subsequently disassembled, cleaned and reassembled, along with correcting an unrelated coincidental electrical outlet failure, to restore proper

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Date	Maintenance Description
	function with no further problems noted. (On 16 June 2005, MLP #05-12 was closed.)
28 Jun 05	On June 27, 2005, the reactor was started up at 1553 hours and run at full power starting at 1611 hours using the alternate city water cooling mode to complete the quarterly Radiological Survey of Restricted Areas (Q-5 Surveillance). The Q-5 Surveillance was completed at 1645 hours and the reactor remained at full power to irradiate samples for UCF researchers. Subsequently, a high primary coolant temperature spike at 921.9° F on the SE Fuel Box (Point #1) was received resulting in a blade drop trip at 1653 hours. With all control and safety systems noted to be responding properly, the reactor was secured at 1653 hours. The temperature of 921.9° F on the SE Fuel Box (Point #1) was noted to return to normal again by spiking down at about 1659 hours. Because of the essentially instantaneous >800° F change in temperature on the SE Fuel Box with all other indications normal, this event was evaluated to be due to a failure in the temperature monitoring system rather than to any actual condition requiring a high temperature trip. In particular, the core bulk outlet temperature at Point #8 was noted to be unchanged during the event. Subsequently, the occurrence was evaluated to be due to an intermittent fault in the temperature monitoring system, possibly exacerbated by the ~12° F higher primary coolant exit temperature experienced at full power in the city water cooling mode. Under MLP #05-13, opened to eliminate the computer or thermocouple as the source of the problem, a higher temperature was simulated. Basically, the use of a simulated high temperature (147° F) with no trip proved that the computer/virtual display device works properly. Next, to identify the thermocouple as failed or failing, resistance readings were made on the thermocouple lines for the SE Fuel Box as well as for all other nine (9) points monitored. The results show that, if the SE Fuel Box thermocouple did separate, or any connection to the thermocouple separated, then it has restored itself mechanically. This result indicates the fault is intermittent and difficult to isolate. At this point, UFTR Form SOP-0.6A (Unscheduled Reactor Trip and Evaluation) was completed, MLP #05-13 was closed out, and the reactor approved for restart subject to a successful daily checkout.
	Since this occurrence was from a known cause, this trip is not considered promptly reportable, though an NRC Project Manager was told of the occurrence on June 29, 2005 in relation to other issues. In addition, it is evaluated as having had negligible effect on reactor safety and no effect on the health and safety of

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Date	Maintenance Description
	the public with a successful daily preoperational check completed on June 30, 2005 with no further problems noted at month's end. Completed Form SOP-0.6A (Unscheduled Reactor Trip and Evaluation) is Attachment I to the June 2005 monthly report. (On 28 June 2005, MLP #05-13 was closed.)
11 Jul 05	During performance of the quarterly scram checks, the secondary cooling system flow meter indicator was noted to be sticking. Under MLP #05-14, the meter internals were cleaned on July 11, 2005; when this did not restore proper operation, the meter was disassembled and a replacement vane was ordered on July 12. The new vane was installed on July 19 but the trip function was still not restored as the secondary well pump tripped after a short period of time precluding further checks. At month's end, other concerns are of more importance so work on this problem has been interrupted after restoring the secondary cooling system for city water use on July 19, 2005. (MLP #05-14 remains open.)
22 Jul 05	Following the high temperature trip (spurious) caused by indicated high temperature on the southeast fuel box outlet thermocouple indication which occurred after operating for 42 minutes at 100 kW (72.333kWh) on June 27, 2005, the event recurred after operating at 100 kW for 27 minutes (47.333 kWh) on July 22, 2005. On July 22, 2005, the reactor was started up at 1555 hours and run at full power starting at 1616 hours using the alternate city water cooling mode to complete an update of part of the quarterly Radiological Survey of Restricted Areas (Q-5 Surveillance) following rearrangement of the rabbit system shielding to reduce radiation levels. The reactor was simultaneously being run to conclude irradiation of samples for UCF researchers. The partial Q-5 Surveillance was essentially completed at 1643 hours when a high primary coolant temperature spike at 921.9° F on the SE Fuel Box (Point #1) was again received resulting in a blade drop trip at 1643 hours. With all control and safety systems noted to be responding properly, the reactor was secured at 1644 hours. The temperature of 921.9° F on the SE Fuel Box (Point #1) was noted to return to normal again by spiking down at about 1649 hours. Because of the essentially instantaneous >800° F change in temperature on the SE Fuel Box with all other indications normal, this event was again evaluated to be due to a failure in the temperature monitoring system rather than to any actual condition requiring a high temperature trip, especially since the core bulk outlet temperature at Point #8 was noted to be unchanged during the event. Subsequently, the

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Date	Maintenance Description
	occurrence was again evaluated to be due to an intermittent fault in the temperature monitoring system, possibly exacerbated by the ~12° F higher primary coolant exit temperature experienced at full power in the city water cooling mode. The use of a simulated high temperature (147° F) with no trip proved that the computer/virtual display device works properly following the earlier trip. Following the earlier trip, resistance readings were made on the thermocouple lines for the SE Fuel Box as well as for all other nine (9) points monitored. The results showed that, if the SE Fuel Box thermocouple did separate, or any connection to the thermocouple separated, then it has restored itself mechanically. These results had been taken to indicate that the fault is intermittent and difficult to isolate. Nevertheless, it was noted that the system had returned to normal. (No MLP has been opened since the system is operational.)
25 Jul 05/ 18 Aug 05	During performance of the daily preoperational check on July 25, 2005, the 3-second period trip bulb was discovered to be burned out. Upon replacement, the new bulb was found also to be not indicating. Under MLP #05-15, the K5 (period scram) relay was checked visually and it was noted to cycle properly when console power was removed. It was also noted that a 3-second signal gives a blade trip but without proper indication no period scram light. On July 28, 2005, the test adapter was inserted on the A11 card (period bistable card) and the bistable trip was adjusted for audible relay action with no response. The op-amp was then tested for proper switching against the input signal but did not appear to switch, apparently because the console switch needed to be reset for proper operation. Using the console key, multiple tests were performed and the K5 card was temporarily switched with a spare with no change. With the A11 card on the extender for evaluation, the period scram relay was noted to be operating properly, possibly due to mechanical agitation; however, with the extender card removed, the problem persisted. Next the K5 relay was temporarily switched with a spare since the indicating bulb was failed initially so arcing at the relay points was possible. The blade continued to trip on a 3-second period but without proper indication (no period scram light). Subsequently, the K5 and K4 relays were swapped for testing but again the problem remained so the relays were replaced to original locations. At this point, the problem was isolated to an apparent failure on the A11 card. Replacements were then ordered for the HP11D relay on the card; no replacements were identified for the HGSM5001 wetted relay on the card.

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Date	Maintenance Description
	On July 28, 2005, the HP11D relay was removed and replaced with the newly acquired spare. The A11 card was then replaced and the period trip test performed 4 times successfully. The water level was then raised and Section 7.2.5.5.2 of the daily preoperational checkout (period trip test) was again performed 4 times with all tests satisfactory so the period bistable trip failure was considered repaired. Since this failure of the period bistable trip was discovered during preoperational checks, it is not considered promptly reportable. This event is considered to have had no impact on the health and safety of the public or facility personnel and negligible impact on reactor safety. (On 28 July 2005, MLP #05-15 was closed.) During a subsequent daily checkout on August 18, 2005, the period trip test failed again. MLP #05-15 was reopened on August 18 for troubleshooting and analysis of the intermittently failing period bistable trip. After some testing it was discovered that the semi-conductor signal chain in the A11 bistable card was at fault. Two of the semi-conductors were difficult to obtain (one of which still has not been obtained, the A51 operational amplifier). At this point it was deemed that testing should continue with the available part, so the Q51 equivalent transistor was subsequently replaced on August 19, 2005 under 10 CFR 50.59 Evaluation Number 05-03 (Transistor Change in A11 Board for Failed Period Trip Bistable). After installation, the period bistable trip was checked satisfactorily three times but then it failed two times. It was thought the failures might be occurring due to temperature effects, so several temperature tests were performed on August 22, but these tests were unsuccessful or successful whether at lower temperatures soon after restoring electrical power or after power was restored for a longer period. Though not certain, the apparent failure was thought to be the operational amplifier on the A11 board. Testing showed that the replaced transistor was not at fault but it did provide isolation for the other transistor leaving only the operational amplifier as the failed component. However, as of August 22, no source could be identified for the operational amplifier that could be counted on. The only source found was Dionese Electronics in Argentina which was uncertain. After some further discussion with General Atomics, it was discovered that this card in a later revision was available for ~$3,000. General Atomics sent the card priority overnight on August 23 and the card arrived on August 24, 2005 when it was tested to assure basic operation and compatibility. It was noted that the version of the NT-4 card used at the UFTR requires a 56.2K 1% 1 Watt precision resistor. This resistor

Date	Maintenance Description
	could not be readily obtained on the open market so General Atomics finally provided one through Mouser Electronics but it proved to be the wrong wattage. Caddock Electronics was then contacted on August 26, 2005 and agreed to make the substitute resistor at no cost but since it was being manufactured, it would not be available until August 30. On August 30, 2005 the precision resistor was received (along with spares). The resistor was then installed on the A11 card under 10 CFR 50.59 Evaluation Number 05-04 (Replacement/Upgrade of NT-4 Bistable Card for Period Scram/Fast Period Interlock (A11 Card)) on August 31, 2005 and a series of tests were run to assure proper functioning of the 10 second fast period interlock and the 3 second period scram. These values were aligned and tested satisfactorily with no fewer than 16 tests in both the water-down and water-up/ blade-up configurations. All blades dropped as expected and a successful daily checkout was performed to assure the bistable period trip was operating properly with no further problems noted. Since this failure of the period bistable trip was discovered during preoperational checks, it is not considered promptly reportable. This event is considered to have had no impact on the health and safety of the public or facility personnel and negligible impact on reactor safety. (On 31 August 2005, MLP #05-15 was reclosed.)
2 Aug 05	Following the high temperature trip (spurious) caused by indicated high temperature on the southeast fuel box outlet thermocouple indication which occurred after operating for 42 minutes at 100 kW (72.333kWh) on June 27, 2005, the event recurred after operating at 100 kW for 27 minutes (47.333 kWh) on July 22, 2005. On July 22, 2005, the reactor was started up at 1555 hours and run at full power starting at 1616 hours using the alternate city water cooling mode to complete an update of part of the quarterly Radiological Survey of Restricted Areas (Q-5 Surveillance) following rearrangement of the rabbit system shielding to reduce radiation levels. The reactor was simultaneously being run to conclude irradiation of samples for UCF researchers. The partial Q-5 Surveillance was essentially completed at 1643 hours when a high primary coolant temperature spike at 921.9° F on the SE Fuel Box (Point #1) was again received resulting in a blade drop trip at 1643 hours. With all control and safety systems noted to be responding properly, the reactor was secured at 1644 hours. The temperature of 921.9° F on the SE Fuel Box (Point #1) was noted to return to

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Date	Maintenance Description
	normal again by spiking down at about 1649 hours. Because of the essentially instantaneous >800° F change in temperature on the SE Fuel Box with all other indications normal, this event was again evaluated to be due to a failure in the temperature monitoring system rather than to any actual condition requiring a high temperature trip, especially since the core bulk outlet temperature at Point #8 was noted to be unchanged during the event. Subsequently, the occurrence was again evaluated to be due to an intermittent fault in the temperature monitoring system, possibly exacerbated by the ~12° F higher primary coolant exit temperature experienced at full power in the city water cooling mode. The use of a simulated high temperature (147° F) with no trip proved that the computer/virtual display device works properly following the earlier trip. Following the earlier trip, resistance readings were made on the thermocouple lines for the SE Fuel Box as well as for all other nine (9) points monitored. The results showed that, if the SE Fuel Box thermocouple did separate, or any connection to the thermocouple separated, then it had restored itself mechanically. These results had been taken to indicate that the fault was intermittent and difficult to isolate. Nevertheless, it was noted that the system had returned to normal so no MLP was opened since the system was operational.
	Since this occurrence was from a known cause, this trip is not considered promptly reportable, though an NRC Project Manager was told of the occurrence on July 22, 2005 because an instant SRO examination had been scheduled for July 2526, 2005. In addition, it was evaluated as having had negligible effect on reactor safety and no effect on the health and safety of the public as with the previous occurrence. Therefore, it was decided to consider limited UFTR operations before undertaking further maintenance efforts.
	On July 25, 2005, the RSRS Executive Committee met to review UFTR status and consider possible future operations at limited power levels following the recurrence of the high temperature trip on July 22 after first occurring on June 27, 2005 while operating on lower flow city water secondary cooling resulting in somewhat higher core fuel box outlet water temperatures of 110°–115° F due to unavailability/unreliability of the secondary well water cooling mode. It was noted that these temperatures are well within the allowable operating temperatures for the cooling water. In both cases the SE fuel box was the indicated failure, though not the hottest box, with the temperature instantaneously spiking to indicate 921.9° F and returning to normal in 56 minutes.

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CHRONOLOGICAL TABULATION OF UFTR PREVENTIVE/CORRECTIVE MAINTENANCE

Date	Maintenance Description
	Because the system had returned to normal and remained so, the RSRS Executive Committee was convened where it was requested that several lower power operations be approved (providing all indications remain satisfactory) with power levels limited to 25 kW to preclude reaching the temperature where the trip might be initiated. The Executive Committee evaluated various input information including the possibility that the high temperature trip is the result of a failing thermocouple or electrical connection and clearly is not an actual temperature transient; that all safety systems have responded properly for both trips; that the likelihood is the intermittent failure will recur at increasing frequency; that the intermittent failure may be due to thermocouple/electrical connections in core and is exacerbated by increased temperature; that the approval for further reduced power operations would be limited in time; that there are multiple indicators to provide reliable reactor core status even after a thermocouple/temperature monitoring channel failure; that reactor operations are needed for an SRO-trainee's NRC license examination; and finally, that there may be a need for two operators to facilitate completion of surveillances after maintenance to repair the temperature monitoring channel should a core entry be needed. After extensive discussions, the Executive Committee voted to approve operations to 25 kW but for only one usage for the SRO-trainee's licensing examination (understood to include prior successful performance of the requisite weekly and daily preoperational checkouts) with no further critical/startup operations allowed until further repairs would be implemented. One committee member also recommended eliminating all possible sources of the trip problem external to the core before undertaking the work to unstack the core. The Facility Director was also to discuss this approval with the NRC Project Manager and the NRC Inspector to assure that they were not opposed to this limited operation. This was accomplished effectively in two conversations with the NRC Project Manager who also contacted the NRC License Examiner to assure the license examination for SRO-trainee Matt Berglund was rescheduled for August 12, 2005. Following successful completion of the weekly preoperational checkout on July 25, 2005, and the RSRS Executive Committee agreement to a single low

Date	Maintenance Description
	power run, UFTR Form SOP-0.6A (Unscheduled Reactor Trip Review and Evaluation) was completed and the reactor approved for the low power operation subject to a successful daily preoperational check which was delayed until July 28, 2005 due to an unrelated period trip bistable card failure (see Section II.C.2 of the July 2005 monthly report). The completed UFTR Form SOP-0.6A is Attachment II to the July 2005 monthly report. The Minutes of the RSRS Executive Committee meeting at which the limited restart was approved are Attachment III to the July 2005 monthly report. At the end of July, the reactor had successful weekly and daily preoperational checks completed and was prepared for the low power operation for the license examination which was completed on August 2, 2005. Subsequently, under Maintenance Log Page #05-16, opened after the second trip, and using the half splitting technique, the panel which contains wiring for the thermocouple system was identified in the equipment pit. This panel additionally contains main power lines (but no distribution blocks) for all pit equipment. This arrangement was considered a possible cause of electromagnetic interference and a possible source of the problem with the high temperature indication on the southeast fuel box outlet thermocouple, but it was considered unlikely as the system has operated for a number of years without a problem of the type seen with this high temperature indication. The six in-core thermocouples are all connected to a terminal block within the panel in the pit. These are the only connections on the block. This block has 2 connectors per side, per thermocouple, with 2 sides and therefore contains 12 total connections. Half of these are constantan and the other half copper. This is consistent with Type T (copper-constantan) thermocouple wiring considerations. Under MLP #05-16, mechanical agitation of the system was begun on August 2, 2005 with light initial agitation defined as slowly moving the individual cables as lightly as possible without interfering with the other cables. The proximity of the incoming (from console) and outgoing (to core) lines prevented perfect isolation of the individual lines. Moving one line could (and in some cases did) contact the other lines regardless of the care with which the chore was performed. The result of the agitation was to generate identical and/or similar temperature spikes on several of the temperature monitoring points including the southeast fuel box

TABLE V-2

CHRONOLOGICAL TABULATION OF UFTR PREVENTIVE/CORRECTIVE MAINTENANCE

Date	Maintenance Description
	outlet (Point #3) and even to a lesser extent on the secondary outlet temperature (Point #10). These results were obtained over the course of several hours, and in some cases were very repeatable. All signal anomalies occurred more than one time. Some anomalies were easily repeated with the same wire motion, others were not. These anomalies were representative of the initial fault which was seen on the southeast fuel box outlet including a spike up and resultant but delayed return to normal. The secondary outlet could not be duplicated. Commonalities considered include the type and age of wire, connection types, connection locations, and path to that location. A general concern at this time was a faulty connection at the block or failure in the wire (hairline fracture) possibly from manufacturing. Subsequently, on August 2, 2005, all connections were reseated (detached and reattached) on the connection block. On August 3, 2005, using a heat gun, the junction box and surrounding wires were heated for 30 minutes. No obvious change occurred during that period, but concern for the insulation necessitated removal of the heat source. A more gradual heat source (1500 W max space heater) was identified and this was used to heat and maintain heat to the area for approximately 2 hours at nearly 90 °F. This is not as hot as the system would probably get during full power operations, especially using city water secondary cooling as was being used at the time of both trips on June 27 and July 22, 2005. During this period no obvious signal changes occurred. Only with mechanical agitation could any changes be induced and these were much reduced, apparently due to the reseating on the connection block completed on August 2, 2005. A substandard mechanism of attachment for several of the spade terminal lugs-to-wire connections was found. The wires were attached by mechanical means employing only a hook through a hole which does not provide a permanent electrical connection. It is possible that the wire, with little heating, could be moving out of contact with the spade lug. It was recommended that these junctions be soldered and the system retested. Zinc chloride flux or equivalent was required to solder the constantan joints.

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CHRONOLOGICAL TABULATION OF UFTR PREVENTIVE/CORRECTIVE MAINTENANCE

Date	Maintenance Description
	At this point the RSRS Executive Committee was convened again, on August 9, 2005. They were requested, upon completion of the soldering and successful retesting, to approve reactor operation at full power with either form of secondary water cooling to verify successful maintenance. After extensive discussions, the Executive Committee voted to approve the 100 kW test run. The resoldering operation was finally completed on August 17, 2005. In addition, thermocouple wire was located and ordered but no replacement thermocouples were located. After August 18, 2005, maintenance time was spent addressing the period bistable failure (see Section II.C.2). The completed UFTR Form SOP-0.6A for the July 22, 2005 trip is Attachment I to the July 2005 monthly report. The Minutes of the August 9, 2005 RSRS Executive Committee meeting at which the limited restart following determination of temperature monitoring wires was approved constitute Attachment I to the August 2005 monthly report which includes the memorandum which outlined the proposed maintenance and subsequent reactor operation to verify operability. At month's end, the reactor was nearly ready for the power run to check repair of the thermocouple monitoring point connections. (MLP #05-16 Remains Open.)
4 Aug 05	Certain parts of the UFTR security system had been periodically difficult to operate. Under MLP #05-17, this portion of the security system is undergoing maintenance and perhaps replacement at month's end with details withheld from public disclosure. (MLP #05-17 remains open.)
5 Aug 05	For some time the secondary well pump would occasionally trip off line after running for some time. On August 5, 2005, the pump began to trip after only about 5 minutes operation. Under MLP #05-18, the well pump system was checked but there was no obvious problem. Gainesville Pump Inc. was contacted. Steve Briel, Manager/technician for Gainesville Pump visited on August 5 and scoped out the system noting several possible sources of the problem but definitely a fault at the well. On August 9, Mr. Denny Macaluso, President of the company and Steve Briel pulled the pump and casing and determined that the pump and motor were failed. On August 1112, a new pump and casing were installed and the pump test run for 2½ hours successfully. Subsequently, PPD was contacted on August 12 relative to relocating/upgrading the external well wiring with PPD's Joe Ellis visiting on August 17 to provide a cost estimate for this work under MWO #0874700. Subsequently it was decided on August 31 to have a better/larger relief valve installed on the well pump

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CHRONOLOGICAL TABULATION OF UFTR PREVENTIVE/CORRECTIVE MAINTENANCE

Date	Maintenance Description
	system by Gainesville Pump Inc. At month's end, a modification package is in preparation to address the electrical upgrade by PPD and the relief valve upgrade by Gainesville Pump. At month's end, the well pump system may be operated for checks but operability to meet Tech Spec requirements awaits installation of the relief valve. (MLP #05-18 remains open.)
12 Aug 05	During efforts to complete reconnecting/soldering the core temperature sensing/ thermocouple lines in the equipment pit, the heat exchanger secondary side sample line was damaged by accidentally bumping it resulting in a sheared connection. Because of water in the pit, the Reactor Manager and Radiation Control Officer were immediately notified through no contamination was expected. Under MLP #05-19 precautionary swipes were taken to document no contamination. Subsequently the threads were cleaned and the coupling replaced on the sample line on the secondary side of the heat exchanger to restore proper operation of the secondary heat exchanger sample connection and the pit was drained of all water with no contamination verified. This event was evaluated to have no effect on health and safety of the public or facility personnel and no impact on reactor safety. (On 15 August 2005, MLP #05-19 was closed.)

MLP #94-14 remains open from 16 March 1994 (Replacement ARM System).
MLP #96-30 remains open from 11 November 1996 (Rechargeable Batteries).
MLP #02-26 remains open from 14 October 2002 (Portable Nimbin SCA and Timer/Counter Modules).
MLP #05-14 remains open from 11 July 2005 (Secondary Well Water Flow Meter).
MLP #05-16 remains open from 2 August 2005 (Temperature Monitoring Point #3 Trip).
MLP #05-17 remains open from 4 August 2005 (Security System).
MLP #05-18 remains open from 5 August 2005 (Deep Well Pump Replacement and System Upgrade).

VI. CHANGES TO TECHNICAL SPECIFICATIONS, SAFETY ANALYSIS REPORT, STANDARD OPERATING PROCEDURES AND OTHER KEY DOCUMENTS

This chapter contains a narrative description and status report on the various changes to key UFTR license-related documents that occurred during the 2004–5 reporting year. As such, this chapter provides a ready reference for the status of various license-related documents to include Technical Specifications, Safety Analysis Report, Standard Operating Procedures, Emergency Plan, Security Response Plan, Reactor Operator Requalification and Recertification Training Program, HEU-to-LEU Conversion Documents as well as Quality Assurance Program Approval for Radioactive Material Shipments and other key documents as they are generated or changed.

A. Changes to Technical Specifications

Technical Specifications Amendment 23 to request that the biennial fuel inspections (B-2 Surveillance) be on a five-year interval like the control blade drive system inspection (V-1 Surveillance) to reduce core entries, decrease likelihood of fuel mechanical damage and better follow ALARA principles was developed and was discussed several times with the NRC Project Manager. It was then reviewed and approved by the Reactor Safety Review Subcommittee on November 8, 2001 and then faxed to the Project Manager on November 8, 2001 and submitted to NRC on November 16. After a round of questions, the facility was informed on December 28, 2001 that the amendment was approved and should be dated December 28 and to contact the Project Manager in the New Year to get a copy. A faxed copy was received on January 3, 2002; the two approved changed pages were then inserted into the console copy of the SOP Manual as approved prior to reactor startup on January 4, 2002. The full original of the NRC approval with Tech Spec Amendment 23 package of pages 19 and 21 dated December 28, 2001 was received on January 7, 2002. This package with the two revised pages marked to agree with facility Tech Spec page markings is included in Appendix A of the 2001–2 report as distributed to all document manuals in early February 2002. There were no requests to change technical specifications during the 2002–3 or 2003–4 reporting years.

There was one change requested in the 2004–5 reporting year. After review and approval at the September 16, 2004 RSRS meeting, proposed Amendment 24 to the UFTR Technical Specifications (R-56 License) affecting pages 19 and 21 was submitted to the NRC with a letter dated September 17, 2004. Basically, this amendment requested that the fuel inspection interval and the interval for mechanical integrity inspection of the control blades and drive system be extended to ten (10) years at intervals not to exceed twelve (12) years from the existing specification of five (5) years at intervals not to exceed six (6) years.

By making the requested change to allow a 10-year surveillance interval on the fuel, the required interval for the surveillance on the fuel per Tech Spec 4.2.7(1) was to match the requested interval for the surveillance on the reactor control and safety system per Tech Spec 4.4.2(4). As a further benefit to reducing dose commitment, these changes will mean these two surveillances can be performed together, further reducing the number of times the core region needs to be entered.

Therefore, this change is well considered to reduce fuel handling and attendant hazards, to reduce the potential for mechanical damage in returning fuel to the core, to reduce the time when the incore fuel is less well protected, and to minimize dose commitment for ALARA considerations—all while optimizing facility utilization and availability.

Since the UFTR facility has been in line for Department of Energy supported conversion from high enriched uranium (HEU) to low enriched uranium (LEU) fuel, going to a 10-year inspection interval for the fuel and mechanical integrity check on the reactor control and safety system was expected to have little effect on this core. In effect, this change is expected to permit the core entry for the two surveillances to be delayed until the fuel conversion is made. At that point, the existing HEU fuel will be removed and fresh LEU fuel added. With removal of all fuel for conversion, the inspection of the control blades and drive systems inside the biological shielding will be facilitated with much reduced dose commitment. Subsequently, following addition of the fresh LEU fuel, the need for inspection of incore fuel elements will be even less justified.

This change as requested was considered to have minor safety significance but large significance for protecting fuel integrity and consistency with ALARA considerations. The entire submittal package is Attachment V to the September monthly report. Although the NRC Project Manager did indicate it was under review at the TRTR meeting in mid-October, there had been no response from NRC on this request as of the end of November 2004. On December 14, 2004, the Project Manager called back and indicated he had only looked at the request and discussed it generally. He indicated that, due to the security workload, such requests were only getting addressed if they impacted or interrupted operations. He finally indicated that it would probably be the second week of January 2005 before it was completed. Subsequently the Project Manager questioned the meaning of the wording in one paragraph as indicating we were dropping the control blade inspection surveillance. After accepting the explained intent, he asked that it be explained and the analysis "updated." After another discussion on December 14, the decision was made to rewrite the first full paragraph of the analysis on page 2 of the original submittal. The Project Manager then indicated that he could proceed with the Safety Evaluation work but that the change should be sent to overnight FedEx and he would docket it. The Tech Spec Amendment 24 update letter clarifying the original request is dated December 15, 2004 and was sent by FedEx to be received on December 16, 2004. The letter is Attachment V to the December 2004 report.

The NRC Project Manager was contacted on January 5, 2005 and asked about the status of the amendment. Subsequently, he indicated that it had been finally signed off on January 4 and could be used as it had been approved exactly as requested. He emailed a copy of the approval package on January 6, 2005. The package was verified on January 7, 2005 with the original received on January 10, 2005. Subsequently, the copies of Tech Spec Amendment 24 were prepared for distribution and implemented to close out this effort. A copy of the complete approval package including the cover letter from NRC, the license amendment itself as approved and the attached NRC Safety Evaluation constitutes Attachment II to the January 2005 monthly report and is contained in **Appendix A** of this report.

B. Revisions to UFTR Final Safety Analysis Report (Relicensing Documentation)

The requirements for renewal of the R-56 operating license were communicated by letter dated May 3, 2002 and received on May 13, 2002. A copy of the letter is Attachment V to the May 2002 monthly report as this set of documents had to be received by NRC at least 30 days before the current license expires on August 30, 2002 in order for the license to remain effective during the relicensing review process which could require several years. The entire relicensing package was submitted to the NRC Document Control Desk with a copy to NRC Region II offices under cover letter dated July 29, 2002. This cover letter is Attachment VII to the July 2002 monthly report. The contents of the package included the following items:

- *Letter of Application* for relicensing per 10 CFR 2.104, signed by the NRE Chairman, the Dean of the College of Engineering, and the University Provost which is Attachment VIII to the July 2002 monthly report.

- Updated *Safety Analysis Report* (original and 10 copies) following the NUREG-1537 format which includes financial qualifications, environmental report information and technical specifications in the applicable portions of the report.

- Updated *Technical Specifications* (1 copy) with a separate cover letter to explain the major changes in the tech specs aside from simple reformatting and reorganization into standard form which involved a complete rewriting of the tech specs. The separate cover letter is Attachment IX to the July 2002 monthly report.

- Updated *Emergency Plan* (original) with a separate cover letter to explain changes which are relatively minor and related to changes in the Tech Specs. The separate cover letter is Attachment X to the July 2002 monthly report noting this would be proposed Revision 13 of the Emergency Plan.

- Updated *Operator Requalification and Recertification Training Program Plan* (1 copy) with a separate cover letter to explain minor changes which are again related to changes in the tech specs. The separate cover letter is Attachment XI to the July 2002 monthly report.

No documentation was included in the package for the Physical Security Plan since an approved PSP for the UFTR is on file with the NRC. The intent is that the NRC will use the existing approved security plan to support the application to relicense the UFTR.

Verification that the submittal was received to meet the application deadline for relicensing per 10 CFR 2.104 to keep the UFTR licensed during the extensive review process was made in a telephone call from the NRC Project Manager on July 31, 2002. By letter dated August 16 and received on August 26, the facility was officially notified that NRC acknowledges receipt of the application dated July 29, 2002. Furthermore, the letter states, "Since your application has been submitted at least 30 days prior to the expiration date of your license, you have satisfied the requirements of 10 CFR Part 2, Section 2.109 (10 CFR 2.109), entitled, 'Effect of Timely Renewal Application.' Accordingly, pursuant to 10 CFR 2.109, the existing license will be deemed not to have expired until the request for renewal has been finally determined." Since the letter clearly referred to the UFTR but incorrectly referenced Operating License R-130 versus R-56, the NRC Project Manager was contacted on August 27 and indicated the letter is only a courtesy and not required so the license number error is not important and the UFTR license will remain in effect past August 30, 2002. The letter acknowledging the UFTR license renewal application is Attachment III to the August 2002 monthly report.

Because of the size of this submittal, the various documents are on file and available as allowed at the facility. The letter of application for relicensing and the NRC letter of acknowledgement of receipt are contained in Appendix B of the 2001–2 annual report. After submittal some errors were noted, primarily due to computer formatting and retrieval errors made during the document conversion process for duplication (printing) of the Final Safety Analysis Report (FSAR). There were no actual changes to the FSAR content or analysis so these changed pages were provided to NRC with a cover letter dated February 23, 2003. As allowed, this package as submitted to NRC is available for review at the UFTR facility.

There had been no comment from NRC though the NRC Project Manager indicated on August 14, 2003 at the TRTR meeting in Hamilton, Ontario that review of the submittal had begun. There had been no subsequent communication on the submittal until the Project Manager indicated at the October 2004 TRTR meeting that questions were being formulated.

With the renewed DOE commitment for HEU-to-LEU conversion, the NRC indicated on March 4, 2005 that they were getting near to issuing the Request for Additional Information (RAI) on the relicensing submittal. However, they will delay relicensing considerations for the HEU-to-LEU conversion effort. Basically, the Project Manager indicated that Chapter 13 of NUREG 1537 will be useful for the Maximum Hypothetical Accident, as will the NUREG on Argonaut accidents which should also be consulted. He did indicate the MHA-related question and some consideration of our safety limit for power level will have to be addressed as part of the HEU to LEU license amendment. The NRC will allow the remainder of the questions to await the HEU/LEU conversion. The RAI was finally received on April 11, 2005 with a cover letter dated April 5, 2005. The letter notes that NRC is continuing its review of the submittal and that questions have arisen for which additional information and clarification is required. However, because of the recent DOE decision to move forward on the HEU-to-LEU conversion, the NRC staff will discuss the timing of responses to the questions. The complete RAI package including the letter from NRC Senior Project Manager Al Adams and the enclosed listing of 92 questions is Attachment IV to the April 2005 report with a relatively few questions needing to be answered for the HEU-to-LEU conversion license amendment. In May, NRE senior Garry Joseph began a project to develop answers to selected questions as a class project (ENU-4905) with some progress occurring, but few answers completed at year's end. The entire RAI package is contained in **Appendix B** of this report.

There have been no other subsequent revisions of the UFTR FSAR. However, with completion of most neutronics and thermal-hydraulics analyses to support the HEU-to-LEU conversion, other FSAR updates are planned as necessary to keep the FSAR current and to support the planned HEU-to-LEU fuel conversion and subsequent preparations for relicensing the UFTR.

C. Generation of New Standard Operating Procedures

One new Standard Operating Procedure (SOP) was generated during the 1999–2000 reporting year but no new SOPs were generated during the 2000–2001, 2001–2 or 2002–3 reporting years. This condition marks the maturity of the UFTR Standard Operating Procedures as great efforts have been undertaken to implement good practice requirements in generating new procedures. During the 2003–4 reporting year, two more new procedures were generated. During the 2004–5 reporting year one additional procedure was generated with title as follows:

- UFTR SOP-F.10, "Background Investigations"

This procedure was generated for better control of facility activities and is withheld from public disclosure.

D. Revisions to Standard Operating Procedures

All existing UFTR Standard Operating Procedures were reviewed and rewritten into a standard format during the 1982–83 reporting period as required by a commitment to NRC following an inspection during that year. As committed to NRC, the final approved version of each SOP (except certain security response procedures which are handled separately) is permanently stored in a word processor to facilitate revisions and updates which are incorporated on a continuing basis in the standard format.

Table VI-1 contains a complete list of the approved UFTR Standard Operating Procedures as they existed at the end of the previous (2003–4) reporting year exclusive of applicable Temporary Change Notices (TCNs) since these do not change procedure intent. Table VI-2 contains a similar complete up-to-date list of the approved Standard Operating Procedures as they exist at the end of the current (2004–5) reporting year. The latest revision number and date for each non-security (not withheld from public disclosure) related procedure is listed in Table VI-2 in parentheses for each SOP; TCNs refer to minor changes made to an SOP in lieu of a full revision and are not noted on the two tables to simplify the presentation. A comparison of Tables VI-1 and VI-2 indicates that there were no revisions to SOPs generated during this reporting year. The most common reasons for SOP revisions are to update minor inconsistencies, correct typographical errors, clarify intent, collect all previous TCNs, etc. Few revisions involve any substantial change in procedural intent—most are intended to clean up the procedure in question, usually as a result of the biennial evaluation of procedures (B-4 Surveillance), as were all the revisions in the 2003–4 reporting year, and, in some cases, simply to update the computer medium/format of storage for the procedure

In previous reporting years, twenty-nine TCNs were issued in 1995–96, eleven in 1996–97, eight in 1997–98, fifteen in 1998–99, twenty in 1999–2000, nine in 2000–2001, twenty in 2001–2, and four in 2002–3 to correct minor discrepancies or better express the unchanged intent of different procedures. In the 2003–4 reporting year, thirteen TCNs were issued for eleven different SOPs. In the 2004–5 reporting year only three TCNs were issued for SOPs including SOP-0.5, E.4 and E.6. It should be noted that the TCNs usually affect only one page, or at most a few pages. When more pages are affected, a revision is usually generated.

As noted above, the TCNs involve minor changes affecting one or a few sections of the respective SOP, sometimes as little as a single sentence. All were fully reviewed by UFTR facility management and approved by the RSRS. Because of the quantity of paper involved and the relatively minor nature of TCNs and even the revisions, copies of these SOP changes or the SOPs as currently revised and implemented are not included in this report. A copy of each may, however, be obtained directly from the UFTR facility if desired.

E. Revisions to UFTR Emergency Plan

With a letter dated August 13, 2001, Revision 12 to the approved UFTR Emergency Plan was submitted to the NRC on August 20, 2001. Revision 12 was reviewed by UFTR management and the Reactor Safety Review Subcommittee (RSRS) to assure Revision 12 does not decrease the effectiveness of the UFTR Emergency Plan. All the changes are considered relatively minor in nature; they are the result of reviews of the Plan and our plans for and responses to simulated emergencies. Most are simple changes to account for name changes or correct typographical errors.

Revision 12 consists of a set of updates and revisions to eleven (11) pages: title page, v, 1-6, 1-11, 5-1, 7-3, 8-1, 8-2, 8-3, 8-4, and 8-5, as well as Appendix II – Agreement Letters. The new pages are marked with the usual vertical lines in the right margin for easy location of specific changes.

All these changes had been reviewed by UFTR management and by the Reactor Safety Review Subcommittee to assure they did not decrease the effectiveness of the UFTR Emergency Plan. In general, these changes make the Plan better suited to assure a proper response to emergencies at the University of Florida Training Reactor. A copy of the complete submittal is Attachment III to the August 2001 report and is contained in Appendix C of the 2001–2 annual report.

With a letter dated January 29, 2002 and received on February 4 the NRC acknowledged receipt of the letter dated August 13, 2001 which transmitted Revision 12 changes to the Emergency Plan for the University of Florida Training Reactor. The NRC letter notes that based on our determination that the changes do not decrease the overall effectiveness of our Emergency Plan, NRC approval is not required. The letter also notes that the initial screening of these changes using NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors," indicates them to be in accordance with 10 CFR 50.54(q) and that our plan continues to meet the requirements of Appendix E to 10 CFR Part 50. Therefore, implementation of these changes would be subject to inspection to confirm that they did not decrease the effectiveness of our Emergency Plan. A copy of this letter is Attachment IV to the February 2002 monthly report. Subsequently, with a distribution memorandum dated February 11, 2002, the changes were distributed internally to be inserted in facility copies of the Emergency Plan and externally to all holders of the Emergency Plan to implement this change fully. All facility copies of the Emergency Plan were updated by February 14, 2002 to implement fully Revision 12.

There were no further revisions of the Emergency Plan generated during the 2002–3, 2003–4 or 2004–5 reporting years.

F. Revisions to UFTR Physical Security Plan

In the 1994–95 reporting year, as a result of a Safeguards and Material Control and Accountability Inspection conducted by NRC inspectors on May 18–19, 1995, several recommendations were made including submitting a Security Plan change concerning material allowed on site. They also reviewed a security plan procedure change identified by UFTR review and outlined the proper submission procedure. No violations were identified. With a letter dated July 18, 1995, Physical Security Plan Revision 12 was submitted to NRC as promised to the NRC inspectors. As indicated to the inspection team, this revision involved one change to the plan concerning allowable quantities and locations for special nuclear material on site as well as one correction of a section number in SOP-F.2. In addition, one further minor change was submitted to update SOP-F.2. Since these changes involved no reduction in the effectiveness of the Security Plan, they were submitted per 10 CFR 50.54(p) to keep the Plan updated. The NRC requested and additional information was submitted by letter dated October 27, 1995 and the revision was finally approved by letter dated November 2, 1995. This revision is withheld from public disclosure.

As a result of the annual RSRS audit and a review for training, Physical Security Plan Revision 13 was submitted to NRC per 10 CFR 50.54(p) with a letter dated June 6, 1996 to update various sections of the Security Plan to correct typographical errors, name changes, errors in the text and a number of inconsistencies in the Security Plan, all of which were considered minor in nature. Subsequently, this revision was approved by letter from NRC dated June 19, 1996. This revision is also withheld from public disclosure.

As a result of conducting the Biennial Evaluation of the UFTR Standard Operating Procedures (B-4 Surveillance) completed near the end of the 1996–97 reporting year, Temporary Change Notices were generated and approved for six security response procedures per Table VI-3. The procedures are withheld from public disclosure and are part of the UFTR Physical Security Plan. Changes involved primarily updating the procedures for the name change to the Nuclear and Radiological Engineering Department and movement of all UFTR inspection and reporting requirements from NRC Region II to NRC Headquarters. As a result, Revision 14 of the UFTR Physical Security Plan was under development at the end of the 1996–97 reporting year for submission in the 1997–98 reporting year.

Physical Security Plan Revision 14 was finally submitted to NRC on October 9, 1997 via letter dated October 7, 1997 referencing an attached letter dated September 25, 1997 describing changes and attached change pages submitted per 10 CFR 50.54(p). Most of the changes were administrative in nature such as updating the Plan for changes in the name of the department from "Nuclear Engineering Sciences" to "Nuclear and Radiological Engineering," updating the name of the Radiation Control Office to the Environmental Health and Safety Division, Radiation Control and Radiological Services Department, and changing written submissions to reflect that regulation of non-power reactors is now from the NRC Non-Power Reactor Directorate office and not Region II per a letter from Luis A. Reyes, Region II Regional Administrator dated August 1, 1997 and communications with Project Managers Marvin Mendonca and Ted Michaels at the Non-Power Reactor Directorate. The cover page is Attachment III to the October 1997 facility monthly report. There had been no response from NRC; however, NRC inspector Stephen Holmes indicated on October 8, 1998 that no approval would be given for changes reviewed by the licensee as not reducing Security Plan effectiveness per 10 CFR 50.54(p). Therefore, the changes were incorporated into the Security Plan on October 23/26, 1998 to close out implementation of Revision 14 which was the last revision implemented.

No further changes have been requested though a number of so-called compensatory measures have been and continue to be generated and/or are under consideration as a result of NRC efforts to address heightened security concerns.

G. Biennial Reactor Operator Requalification and Recertification Program

The existing operator requalification and recertification program training cycle for the University of Florida Training Reactor as submitted with a letter dated June 6, 2003 was scheduled to end in June 2005. Therefore, it was proposed to renew the current plan with minor changes. The revised plan is essentially the same as that currently being used for the two-year training cycle except for date changes. A copy of this renewed plan was submitted to NRC on May 28, 2005 with a letter dated May 25, 2005. The renewed plan will cover the UFTR operator requalification and recertification program from July 2005 through June 2007. As indicated in the letter to NRC, the UFTR facility plans to continue using this proposed program beyond the next two-year cycle; that is, the facility will automatically restart the same two-year requalification and recertification program training cycle every two years until the revised plan submitted with the relicensing application is approved. Though not formally reviewed the NRC Project Manager indicated that the proposed changes meet the applicable requirements of 10 CFR 55 and are acceptable subject to inspection. The complete submission to NRC is contained in **Appendix C** of this report.

H. UFTR ALARA Program

As the part of the process of implementing the requirements of the new 10 CFR Part 20, a UFTR ALARA Program was generated. This ALARA Program was developed to be consistent with the University of Florida ALARA Program as well and was implemented along with the new 10 CFR Part 20 in January 1994. A copy of the original UFTR ALARA Program was in Appendix D of the 1993–94 annual report. This ALARA Program was updated via Revision 1 in August 2002 to remain consistent with the University Program. Though the changes are considered minor, a copy of the revised ALARA Program was contained in Appendix D of the 2001–2 annual report with no changes occurring in the 2002–3, 2003–4 or 2004–5 reporting years.

I. UFTR Respiratory Protection Program

NRC Inspection Report No. 50-83/94-01 dated April 6, 1994 contained a Severity Level IV Notice of Violation for the failure to have issued a written policy statement on respirator usage and for not having advised users that they could leave an area at any time for relief. Also, the potential respirator users had not been fit tested for the types of respiratory protection equipment at the facility. During May 1994 much work was performed on developing the required respiratory protection program. The facility reply to the Notice of Violation was submitted to NRC as a letter dated May 6, 1994. It indicated that a written statement to all potential respirator users informing them that they may leave the area at any time for relief was issued on May 2, 1994 and that the written policy statement concerning respirator usage was under development with full compliance including documented review and approval of the policy committed to be achieved by August 31, 1994. In a letter dated May 25, 1994 and received on May 31, 1994, the NRC indicated that they had evaluated the UFTR response and found it met the requirements of 10 CFR 20.201 [should be 20.2001].

A draft Respiratory Protection Program was completed and submitted to the RSRS on August 25, 1994. The NRC (Craig Bassett) was informed that the Program would not be approved by the August 31, 1994 commitment date and indicated that such should be officially transmitted to NRC. Subsequently, via letter dated August 31, 1994, the delay in the UFTR commitment was transmitted to the NRC with a new commitment to have the UFTR Respiratory Protection Program approved at the next RSRS meeting scheduled for September 29, 1994 and full compliance including documented review and approval of the policy achieved by September 30, 1994. The initial revised version of the Respiratory Protection Program with a Policy Statement was finally reviewed and approved by the RSRS at its meeting on September 29, 1994 and implemented on September 30, 1994. A revised UFTR Respiratory Protection Program (Revision 1) amending the required frequency of medical examinations was implemented on March 16, 1995. The original (Revision 0) Program Document as well as the Revision 1 version of the UFTR Respiratory Protection Program are contained in Appendix E of the 1994–95 annual report. The Severity Level IV Notice of Violation for failure to comply with all portions of the Respiratory Protection Program was finally closed out during the NRC Inspection conducted on May 22, 1996 per page 7 of NRC Inspection Report No. 50-83/96-01.

As a result of core area maintenance, disassembly and inspection efforts in response to a reactivity anomaly, at the end of June 1998 and throughout the month of July, efforts were under taken to modify the approved UFTR Respiratory Protection Program to allow use of half respirator masks and to schedule the necessary medical examinations for which there was some delay. The necessary physicals for two individuals were conducted on 10 July 1998. The revised UFTR Respiratory Protection Program was ready for internal review and approval by 24 July 1998 but the RSRS Executive Committee was unable to meet for several days. On 24 July 1998, NRC Senior Project Manager Ted Michaels was updated on the status of the checks on the reactivity problem including probable separation on one control blade and plans to disassemble the entire core since borescope indications are somewhat limited. He was also informed of the detection of airborne particulates at low levels and stop of work and delays in developing and approving the revised Respiratory Protection Program. Specifically, we discussed the use of half-face respirators, status of exams/physicals, etc., and 10 CFR 20.1703(d) requiring notification of the Region II Administrator 30 days before the date of using respiratory protection equipment the first time. Since we normally go directly to the NPR Directorate, we requested direction on what to do next. He was not sure whether we should send in something and asked that he be contacted again on July 28 which was done, whereupon he indicated we should send in the proposed Program when internally approved. Revision 2 of the UFTR Respiratory Protection Program was finally internally approved along with the proposed Policy Statement at an RSRS Executive Committee meeting on July 30, 1998. Subsequently, NRC Senior Project Manager Ted Michaels was contacted on July 30 and he requested submission of the Program for review indicating it should not require 30 days. The internally approved Respiratory Protection Program Revision 2 and the proposed Policy Statement were faxed to the Project Manager on July 30, 1998 to get the review started with the formal submission by letter to the Document Control Desk then accomplished on August 3, 1998.

At the beginning of August, maintenance operations were awaiting NRC review of the Respiratory Protection Program Revision 2. On August 3, 1998, NRC Inspector Stephen Holmes of the Non-Power Reactor Directorate indicated he would visit for an inspection on August 13–14, 1998 in order to provide on-site review verifying that the Respiratory Protection Program Revision 2 was acceptable and reviewed by NRC prior to implementation. Therefore, all the preliminary aspects of implementing the Respiratory Protection Program Revision 2 were addressed prior to his arrival to include acquiring half-face respirators and arranging a visit by Mary Russell on August 6 to provide half-face respirator fits and training three personnel. Subsequently, Vince McLeod provided the same fit tests and training for two other operations personnel including the Facility Director with the whole Respiratory Protection Program Revision 2 administratively reviewed and all documentation completed prior to Mr. Holmes arrival. Upon his arrival on August 13, Mr. Holmes toured the facility to check on maintenance status, he checked records of fit testing and training as well as the Program itself. Though he continued to interview personnel and check the fit testing equipment on August 14, Mr. Holmes evaluated that the Program was ready for implementation on the afternoon of August 13, 1998. Therefore, the official implementing memorandum for the Program was issued on August 13, 1998. A new Radiation Work Permit 98-8-I was also opened allowing use of respirators per the Respiratory Protection Program Revision 2 and requiring SRO supervision of operations among other controls with respirators used for moving graphite on the afternoon of August 13with observation by Mr. Holmes. Inspector Holmes held his exit interview on August 14 prior to leaving indicating no problems were identified and respirators are not required but are optional at the worker's convenience. Subsequently, more graphite was removed on the afternoon of August 14 which was the last day that workers opted to wear respirators as airborne radioactivity levels were measured to be quite low. Subsequently, the RWP 98-8-I was reissued several times during the month as work progressed slowly on further disassembly of the reactor core to address the reactivity anomaly. These respirators were used only a couple of times as airborne contamination levels were very low. There have been no further changes to the UFTR Respiratory Protection Program in the 1998–99 or any subsequent reporting years.

J. <u>HEU to LEU Fuel Conversion Documents</u>

The original proposal submitted to NRC to meet 10 CFR 50.64 requirements for scheduling UFTR conversion from HEU to LEU fuel was accepted as meeting the legal requirements for submission in March 1987. However, in a letter dated April 17, 1987 and received on April 22, 1987, the NRC claimed the scheduled span of time from receipt of funding to submittal of our application to convert was too long. The updated (reduced) schedule (Revision 1) showing a reduction of 8 months as presented in Table VI-4 was then submitted to NRC licensing in Washington with a cover letter dated May 14, 1987. During subsequent reporting years, new proposals updating the UFTR conversion schedule and work status per 10 CFR 50.64(b)(2) requirements were submitted to NRC each March to meet the annual March 27 deadline.

After receiving funding, work proceeded as quickly as possible though a shortage of graduate students to perform the neutronic and other analyses caused this work to lag each year. In addition, because of extensive efforts to decontaminate and remodel a room in which to store the SPERT LEU fuel, to change the license description of the SPERT storage facility, to move the fuel to the new facility, to release the previous storage room to unrestricted usage, to revise the facility security plan

(SNM-1050) and then to perform a detailed pin by pin visual inspection and verification of serial numbers, the conversion analysis was further delayed in the first two years.

The required visual inspection and identification of SPERT fuel pins was completed on September 19, 1988. As committed, a sufficient number of SPERT fuel pins were radiographed to provide an LEU core and replacement pins for the UFTR by March 31, 1989, when the SPERT usage license was to expire. As for the SNM-1050 License, a significant effort was involved as the renewal license application for renewal under "storage only" conditions was submitted with a letter on March 1, 1989 as required. License No. SNM-1050, as renewed, was dated June 23, 1989 and was received on June 29, 1989. The renewed license authorized "storage only" conditions and has an expiration date of June 30, 1994. The cover letter also specified that any request for amendment to the SNM-1050 License should be submitted in the form of replacement pages to the renewal application submitted on March 1, 1989 with changes or new items clearly identified. Subsequently, in June 1989, an engineering-based decision was finally made not to use the SPERT fuel but rather to use the alternate low enriched silicide plate-type fuel. As a result plans were developed to ship the fuel.

A proposal for support to provide 1200 SPERT fuel pins for transfer for shipment to Oak Ridge National Laboratory was submitted to Martin Marietta Energy Systems, Inc. in January 1990 in response to Request for Proposal CO378-19 dated December 12, 1989. This proposal was submitted to Martin Marietta Energy Systems in January and accepted. Loading of the drums was completed per approved UFSA SOP-U.4 on May 16, 1990 and 1200 pins in 19 DOT type 6M drums plus one (1) empty drum were transferred to Mr. Leon Fair of Martin-Marietta Systems Inc. for shipment by truck to a secure DOE facility at Oak Ridge National Laboratory on May 17, 1990. Revision 3 of the Physical Security Plan (PSP) for the SNM-1050 License was then transmitted to the NRC with a letter dated June 7, 1990 to update the Special Nuclear Material on site following the May 17 transfer of 1200 pins to Martin-Marietta's control. Approval of Revision 3 to the University of Florida SPERT Assembly Physical Security Plan occurred with a letter dated June 20, 1990 and received on June 26, 1990.

An application to amend the storage-only SNM-1050 License to allow storage of the fuel in the North Quonset Hut (Room 6) versus Room 5 of the Nuclear Research Field Building was submitted to NRC with a letter dated June 6, 1990. This SNM-1050 License amendment making the smaller Room 6 an allowed storage location was approved per a letter and license amendment dated June 14, 1990. All of the remaining 4200 SPERT fuel pins not previously shipped were then moved to Room 6 on July 30. Revision 4 of the SNM-1050 Physical Security Plan was submitted to NRC with a letter dated September 13, 1990 while the response to several security allegations was submitted as a letter also dated September 13, 1990. The next security inspection was conducted on October 25, 1990 by NRC Security Inspector Orysia Masnyk, to investigate security violation allegations associated with the SNM-1050 License as well as to consider final approval of Revision 4 to the Physical Security Plan for the SNM-1050 License. In NRC Inspection Report No. 50-83/90-02 dated November 23, 1990, NRC Region II did close out the allegation and accept implementation of Revision 4 of the UFSA Security Plan.

Throughout the 1988–89 reporting year, the neutronics analysis to support the conversion had been progressing at a slow pace with the graduate student involved deciding to leave for another university when not approved to pursue a doctoral degree. This loss greatly hindered analysis work at the beginning of the 1989–90 reporting year. As a result of the overall slow progress on this work related to UFTR HEU-to-LEU conversion and funded by DOE, the proposal submitted to NRC with a letter dated March 22, 1989 to meet the annual March 27, 1989 and 1990 deadlines per 10 CFR 50.64(b)(2) showed a further lengthening of the schedule.

An updated proposal was submitted to NRC with a letter dated March 26, 1991 explaining that a student thesis project had resulted in good progress in assuring neutronics methodology is adequate and the modeling of the existing core was nearly complete lacking only several confirmatory calculations and calculations to predict changes caused by temperature effects. NRC was also updated that only scoping calculations had been completed for the proposed LEU core with the number of fuel plates per bundle not yet set in March 1991. It was expected that DOE-supplied funding support of this work would be extended beyond April 30, 1991 so this work could be concluded along with basic thermal hydraulics analysis to conclude the required HEU to LEU safety analysis. A no-cost extension of the Department of Energy Grant DE-FG05-88ER75387 entitled "Conversion of University of Florida Reactor to Low Enriched Uranium (LEU)" was submitted to Ms. Ann Rydalch via a letter dated April 25, 1991 with a copy supplied to Keith Brown. The extension was agreed to be until April 30, 1992 with notification of the extension not received until fall 1991 making some plans and efforts difficult to implement. The updated proposed schedule submitted as required by March 27, 1991 per 10 CFR 50.64(b)(2) therefore showed a further schedule slippage.

The individual working on the neutronics analysis completed his benchmark calculations on the existing UFTR HEU core in April 1991. Subsequently, he completed his thesis work in May 1991 and continued his work until May 23, 1991. After the number of fuel plates per bundle was set at 14 from the neutronics analysis, thermal hydraulics analyses were begun late in the 1990–91 reporting year. During the 1991–92 reporting year, a graduate assistant continued working on the thermal hydraulics area on the 14-plate fuel bundle arrangement selected for the conversion with good progress made to nearly complete this work during that reporting year. Work on the NRC submission package was also begun with limited progress made. During the 1992–93 reporting year and again in the 1993–94, 1994–95 and 1995–96 reporting years, the delay of official grant extension and unavailability of personnel made financial support of this effort more difficult. The same was true in this latest reporting year, so the latest updated proposal schedule submitted as required on March 27, 1997 per 10 CFR 50.64(b)(2) as Revision 11 therefore shows a further schedule slippage as depicted in Table VI-5 of the 1996–97 report. This further delay is because the basic thermal-hydraulics analysis proceeded more slowly than expected and because of DOE questions about fuel and core design arrangements that are requiring staff time to answer in preparation for approving the final fuel bundle design.

Early in the year, a call was made to Dennis Wilson to have the small remaining DOE-supplied funding support for this HEU to LEU analysis work extended to keep the grant open, but no money is available to support actual conversion as explained in the submittal to NRC and as indicated in a letter from John Gutteridge, Program Director, Office of Planning and Analysis, Office of Nuclear Energy, Science and Technology, dated February 23, 1998 and received in early March 1998. Little was accomplished during this year until October 1997 when visiting Professor Marc Caner from the SOREQ Institute in Israel began working on the project with hopes this project could be concluded this year, since the loss of several facility personnel had prevented work in this area previously. There had been a delay in the response to the grant support extension request to DOE; however, as of the end of January 1998, some DOE money was available to be used to support some of Dr. Caner's work. As required, the 1998 updated proposal on the HEU-to-LEU conversion to meet requirements of 10 CFR 50.64(c)(2) was submitted to the NRC with a letter dated March 27, 1998 again explaining the reasons for delays and indicating the updated proposal for the conversion schedule to include submission of the license amendment safety analysis package is now scheduled for October 1998. However, little was accomplished during the year since the loss of several facility personnel had prevented work in this area, but at year's end Dr. Marc Caner is now spending his sabbatical time since December 1997 on the project and work is progressing though confirming dimensions and materials to support the calculations has involved considerable time during July 1998 with Dr. Caner receiving a tour to observe the unstacked core on August 27, 1998.

During the 1998–99 reporting year, Dr. Caner provided some information on reactivity coefficients and completed his reactor physics analyses for the HEU-to-LEU conversion. A draft copy of his work to date on conversion dated September 23, 1998 was received on September 28, 1998. A "final" copy of his work to date was received on December 16, 1998. During March 1999, the internal review was completed and the report finalized with this work generally agreeing with earlier reactor physics analyses. Several discussions have occurred since as Dr. Caner provided proposed Tech Spec changes in June and left all his work well documented before he finally left on July 20, 1999 to return to the SOREQ Institute.

As required, the 1999 updated proposal on the HEU-to-LEU conversion to meet requirements of 10 CFR 50.64(c)(2) was submitted to the NRC with a letter dated March 29, 1999 again explaining the reasons for delays and indicating the updated proposal for the conversion schedule to include submission of the license amendment safety analysis package would now be scheduled for June 1999. The updated schedule is Attachment I to the March 1999 facility monthly report. Though too late to include in the proposal, a formal letter from John Gutteridge, Program Director, University Programs, in the DOE office of Nuclear Energy, Science and Technology, dated April 7, 1999 and received on April 12, 1999 indicated no conversion funding is available during fiscal year 1999 so there was no need for submission of the HEU-to-LEU conversion document to NRC. The letter is available at the UFTR facility for anyone desiring to examine it.

NRC Project Manager Ted Michaels called on October 15, 1999 to emphasize the need to get the conversion package in within the next few months for proper review. During November 1999, a graduate student indicated interest in working on this submittal for a master's project. During December 1999, she decided to do so as project needs were outlined; she also indicated an interest in doing the license renewal package for her engineer's degree project. In a call on December 2, the NRC Project Manager again emphasized the need to get the conversion package submitted in the next few months.

During January–March 2000, the graduate student began to put the conversion package together though some additional calculations were noted also to be needed for control blade worths and kinetics. In response to a call from Mr. Michaels in March, a message was left that we were preparing the submittal and completing calculations and hoping to get him something by the end of March 2000 but that without DOE funding support, the issue is moot. During April 2000, it was decided the PARET code was needed for kinetics/thermal analysis along with information on control blade geometry both of which were obtained with PARET available by month's end. Access to the NRE storage facility for the previous conversion calculations was not possible due to having the wrong key on April 16. A correct key was ordered and still did not fit in early May 2000 when another key finally accessed the facility to verify no computer output was present. Arrangements were made for the graduate student to have access to an SOP Manual, Tech Specs, Emergency Plan and FSAR on May 19, 2000 and discussions with her on May 31 indicated the CITATION calculations she was to run for control blade worth measurements will require additional funding. Discussions with NRC Project Manager Ted Michaels during a visit to NRC on May 24, 2000 indicated a late summer submission of the HEU to LEU package would be acceptable since fuel is not due before October 2001 and the new federal government fiscal year doesn't start until October 1, 2000. During June 2000, a limited-use computer account was set up for the graduate student with discussions in use of PARET code with a faculty member cognizant of its use and review of some of the package in preparation for NRC submittal. During July 2000, there were several discussions with the graduate student plus partial review of drafts of the NRC submittal package. During August 2000, at the end of the last reporting year, a considerable portion of the submittal was reviewed and discussed as the package was nearing completion.

As required, the 2000 updated proposal on the HEU-to-LEU conversion to meet requirements of 10 CFR 50.64(c)(2) was submitted to the NRC with a letter dated March 29, 2000 again explaining the reasons for delays and indicating the updated proposal for the conversion schedule to include submission of the license amendment safety analysis package which is now scheduled for May 2000. The proposal cover letter and the updated schedule are available for examination at the facility.

Review and discussions of the HEU to LEU submittal package continued in September, October and November 2000 of this reporting year as a number of calculations and checks continued with the package nearly ready for submittal. At the TRTR meeting on October 19, 2000, Mr. Tony Vinnola of DOE indicated there was a possible delay in getting our LEU fuel in late 2001. He suggested we send a letter documenting the expectation to submit the conversion package soon and the desire to receive fuel before the end of 2001. This letter was submitted as required, dated October 24, 2000.

During December 2000, the graduate student successfully defended her project on December 15 so the package is ready for submission to NRC after generation of a cover letter which has not yet been accomplished. During January 2001, she and a fellow graduate student enrolled in ENU-6937 Special Topics in Nuclear and Radiological Engineering Sciences to measure HEU core physics parameters in preparation for conversion. This work was obviously on hold during the extended outage from January 31, 2001 through the end of March 2001.

On March 8 and again March 20, there were discussions with Tony Vinnola of DOE concerning the UFTR HEU-to-LEU conversion. It appears the UFTR fuel may have to be made in two sets if at all. After the March 20 discussion, Mr. Vinnola was to speak with DOE headquarters about UFTR fuel for conversion as we indicated our package was essentially ready for submittal. There has been no word from DOE as there is every likelihood they will not fund our fuel, at least not in the foreseeable future.

With the reactor back up in early April and May 2001, the two students, as part of ENU-6937 – Special Topics in Nuclear and Radiological Engineering Sciences, performed a number of experiments measuring parameters needed for the HEU-to-LEU conversion and/or relicensing. During June 2001, an email was sent to Tony Vinnola at DOE summarizing UFTR HEU-to-LEU conversion considerations. Subsequently, during June there were a number of emails and telephone conversations concerning conversion with Tony Vinnola and DOE headquarters representatives as they are trying to determine plans. No word was received in July 2001 but Tony Vinnola indicated in a conversation on August 15 that Bill Magwood is looking at the cost of HEU-to-LEU conversion versus a replacement HEU core! He was told the cost wouldn't be much different but the regulatory agency might have some concerns. On August 6 an email was sent to Offsite Fuels Receipt Coordinator (SNM) for Westinghouse Savannah River Company at the Savannah River site, indicating no HEU fuel will be shipped from the UFTR before the end of 2002 at the earliest.

As required, the 2002 updated proposal on the HEU-to-LEU conversion to meet requirements of 10 CFR 50.64(c)(2) was submitted to the NRC on March 27, 2002 with a letter dated March 27, 2002 again explaining the reasons for delays and indicating the updated proposal for the conversion schedule to include submission of the license amendment safety analysis package which is now essentially ready for submission pending DOE commitment of support and tentatively scheduled for update in April 2005. The proposal cover letter and the updated schedule are available for examination at the facility.

By email dated July 22, 2002, a DOE DDR Program Manager, transmitted a summary report of fuel assemblies received and projected receipts through 2035 and asked for an update. From the data table, it was not possible to determine if UFTR fuel was included. Therefore, the current UFTR status was communicated indicating that after relicensing submittal, the facility would hope to do an HEU-to-LEU conversion sometime in the not too distant future, probably in 2004. She indicated that they were showing the UFTR shipping 24 assemblies in 2004 and asked if this was correct to which the reply was that it probably was correct as far as we can tell subject to relicensing uncertainty and DOE support. At the TRTR meeting in Salt Lake City on November 12, 2002, a DOE representative asked that he be sent a copy of the UFTR letter requesting relicensing so they would have justification to include the UFTR in new fuel manufacturing plans so a copy of the relicensing request was provided.

As required, the 2003 updated proposal on the HEU-to-LEU conversion to meet requirements of 10 CFR 50.64(c)(2) was not submitted by March 27, 2003 due to an oversight. It was finally submitted to NRC with a letter dated April 3, 2003. This letter contained the usual summary and reasons for delays and indicated the updated proposal for the conversion schedule is dependent upon DOE support. The letter with the proposal notes that the entire package will be assembled for submission to NRC within two months of DOE indicating LEU fuel will be made available with the project progressing as predicted in the enclosed updated proposal. Currently, as noted in the proposal, DOE has indicated there is no money for conversion in fiscal year 2002 (Phase II) and they are not sure about 2003 as they had indicated plans to wait until the UFTR would submit a timely relicensing package for its R-56 license which occurred by letter dated July 29, 2002 in the 2001–2 reporting year. The submittal to NRC is to be prepared and submitted whenever DOE provides the conversion money and subsequently the replacement LEU fuel will be made available, although

DOE has been noncommittal due to budget limitations. Nevertheless, the facility expects to complete a submission within two months of DOE indicating availability of support. The latest proposal cover letter and the updated schedule are Attachment IV to the April 2003 monthly report and are available for examination at the facility.

On February 23, 2004, DOE informed the facility that there may be a further delay in UFTR HEU-to-LEU conversion as some other facilities are being pushed by NRC because of security concerns related to vulnerability assessments. As required, the 2004 updated proposal on the HEU-to-LEU conversion to meet requirements of 10 CFR 50.64(c)(2) was submitted to NRC on March 27 with a letter dated March 26, 2004. This letter contained the usual summary and reasons for delays and indicated the updated proposal for the conversion schedule is dependent upon DOE support. The letter with the proposal notes that the entire package will be assembled for submission to NRC within two months of DOE indicating LEU fuel will be made available with the project progressing as predicted in the enclosed updated proposal. Currently, as noted in the proposal, DOE has indicated there is no money for conversion in fiscal year 2003 (Phase II) and they are not sure about 2004 as they had indicated plans to wait until the UFTR would submit a timely relicensing package for its R-56 license which was done by letter dated July 29, 2002. The submittal to NRC is to be prepared and submitted whenever DOE provides the conversion money and subsequently the replacement LEU fuel will be made available, although DOE has been noncommittal due to recent budget limitations. Nevertheless, it was expected to complete a submission within two months of DOE indicating availability of support. The updated proposal cover letter and schedule are Attachment I to the March 2004 monthly report and are available for examination at the facility.

On August 23, 2004, Tony Vinnola called to update the Idaho/DOE records once again on the DOE fuel the facility has in its possession. His records showed over 100 assemblies so some time was spent getting the information and updating him on actual fuel on site. He indicated that there was renewed DOE interest in conversion but was concerned about estimated conversion costs since the GAO report takes issue with DOE's estimated cost to convert a reactor at over $2 million. With increased shipping costs for an irradiated core and the cost to make a replacement core, such estimates may not be too high. At any rate, Mr. Vinnola discussed timetables relative to UFTR relicensing, getting fuel made, etc. This renewed interest was because of security concerns highlighted by the GAO report and newspaper articles. At the TRTR meeting in October 2004, NRC representatives as well as DOE representatives indicated there is every intention that the UFTR undergo HEU-to-LEU conversion. Again, the concern was availability of funds but with definitely renewed urgency to the conversion.

On February 8, 2005, Tony Vinnola called to say DOE was getting funding from NNSA to convert the UFTR and Texas A&M. DOE's John Gutteridge contacted UFTR Facility Director Bill Vernetson and NRE Department Chair Ali Haghighat on February 11 concerning visiting to discuss HEU/LEU conversion. DOE LEU fuel drawings were verified on February 16 and HEU/LEU fuel issues were discussed with Doug Morrell at INL on February 16–17. A memorandum on UFTR fuel box measured inner dimensions was generated on February 17–18 based on measurements made on the last core entry. Drs. Haghighat and Vernetson met with John Gutteridge (DOE Director, University Programs), Jim Matos (ANL RERTR Program Coordinator of Analysis) and Parrish Staples (DOE Office of Global Nuclear Material Threat Reduction) to include a reactor tour on February 18. The remainder of February was spent generating a statement of work and cost considerations for the conversion. On March 1, 2005, Tony Vinnola was sent a complete copy of the current Tech Specs by Fed Ex for review versus fuel requirements.

On March 16–17, 2005, Doug Morrell, INL Mechanical Engineer, Bill Steinke, INL Nuclear Engineer, Dana Cooper, INL QA Engineer, as well as Tiffany Baxter, BWXT Fuels Engineer and Dave Capp, INL QA Engineer assigned to BWXT, visited and reviewed the fuel specifications for finalizing drawings. On March 22, Mitch Meyer, Manager Fuel Development Section, Nuclear Technology Division, INL, and Dana Meyer, Project Leader, Nuclear Engineering Division, INL, visited to discuss conversion costs and needs with a conference call on March 24 with Parrish Staples, Tony Vinnola, Jim Matos, et al. Work was also begun at month's end to obtain PC resistivity values for the last several years to provide a database relative to needs for treating the LEU fuel.

On the recommendation of EH&S Director W.S. Properzio, Dr. Vernetson was interviewed by Greg Bruno concerning the DOE plans for UFTR conversion on March 24, 2005. The resultant article appeared on the front page of the *Gainesville Sun* on Saturday, March 26, 2005. Though creating quite a stir, the article was generally factual based on information available elsewhere with the exception of the picture caption and the statement that money is available.

As required, the 2005 updated proposal on the HEU-to-LEU conversion to meet requirements of 10 CFR 50.64(c)(2) was submitted to NRC on March 29 with a letter dated March 29, 2005. This letter contained the usual summary and reasons for delays and indicated the updated proposal for the conversion schedule is dependent upon DOE support. The letter with the proposal notes that the entire package will be assembled for submission to NRC within about five months of DOE indicating conversion money will be provided and that subsequently the replacement LEU will be made available. The letter also notes that DOE officials had visited the facility with every indication being that funding could soon be available. The latest proposal cover letter and the updated schedule are Attachment VIII to the March report which is available at the facility.

Review of UFTR LEU Fuel and related drawings was finally completed and the signed Document Management Control System (DMCS) Document Action Request (DAR) form returned to Doug Morrell at INL on April 22, 2005. Various parties began reviewing UFTR drawings of the core to begin developing the reactor physics computational model. During the month a pH meter was procured and measurements were made on the primary coolant; samples of primary coolant were also obtained and resistivity data for the last several years was tabulated. Finally, Jim Wade and others from INL plus Steve Foster and others from Savannah River Lab arranged a visit to the UFTR to discuss considerations for shipping the UFTR irradiated HEU fuel. An April 11, 2005 news release from NSSA indicated that Secretary of Energy S.W. Bodman had announced that the University of Florida reactor along with that at Texas A&M are two research reactors that are planned to be converted to LEU fuel by late 2006 as a "significant step forward to ensure that weapons-usable nuclear material" is less likely to be misused.

During May 2005, efforts continued to develop an analytical model of the UFTR facility for neutronic and thermal hydraulic analysis. NRC Project Manager Al Adams indicated submission of the HEU to LEU license amendment could be in two pieces. Efforts continued to get reactor information for analysis including collecting resistivity data, shipping four graphite samples to INL plus visits to discuss and plan shipment of HEU fuel and receipt of LEU fuel. On May 18, Steve

Foster (Research Reactor Domestic Receipts Technical Lead) and Jim Roach (Fuel Receipt Engineer) from Westinghouse Savannah River Site visited to discuss and review plans to ship HEU fuel with Jim Wade and to review data requirements for doing so. Efforts also continued on the proposal for funding.

During June 2005, HEU-to-LEU conversion efforts continued with a visit by RERTR Program Manager James Matos and Physicist John Stillman to review the core modeling. Various measurements were made to support core neutronics model development necessitating partial rabbit system shielding disassembly and subsequent reassembly and operations at full power to verify adequacy of replaced shielding. Various pieces of information were obtained for the research assistants working on the project. Four primary coolant water samples plus a piece of a dummy aluminum plate were obtained and sent to INL for elemental analysis for boron, etc. During July 2005, support for neutronics calculations continued along with review of several documents including LEU fuel drawings changes and the inconclusive results of the graphite sample analysis. During August 2005, support continued including evaluation of graphite replacement needs and discussions with RERTR program personnel who are performing most of the thermal hydraulics analyses.

K. <u>Quality Assurance Program Approval for Radioactive Material Package</u>

There was no activity since closeout of the SNM-1050 license in the 2001–2 reporting year.

On March 14, 2003, an NRC NMSS representative called to check on the proper contact to send notification that the approved QA Program for Part 71 activities was due to expire on May 31, 2003 so he was updated on the proper contact. The QA Program Approval Expiration Notice dated March 28, 2003 was received on April 3, 2003 and is Attachment V to the April 2003 monthly report and is contained in Appendix B of the 2002–3 annual report.

An NRC NMSS representative called on May 1 to say May 1, 2003 was the last day to apply for automatic extension and requested an email to confirm that he had called and we would not be allowed to perform the program activities after May 31, 2003. A copy of the confirming email is Attachment XI to the May 2003 monthly report. Subsequently, the NRC NMSS representative called again on October 3, 2003 and indicated the QA Program was not canceled; rather, NRC is waiting to renew it. The Facility Director indicated we were in no hurry and had no plans to use the Program and would have to amend it anyway since it was intended for controlling the SPERT fuel shipment which has been completed. The NRC NMSS representative indicated that if it is canceled, a lot of time and effort could be involved to renew it, so it would be better to renew it and then amend it as needed. He indicated a simple renewal letter was all that was needed and provided an email to that effect. After review of the renewal letter by the RSRS on October 23, the renewal request letter dated October 24, 2003 was submitted to NRC. The renewal submission with attached copy of QA Program Approval 0578, Revision 3 is Attachment II to the October 2003 monthly report. The official QA Program renewal dated November 12, 2003 was received on November 17, 2003; it is valid for five years from the previous date of expiration to May 31, 2008. The letter of notification and the enclosed QA Program Approval 0578, Revision 5 is contained in Appendix A of the 2003–4 annual report for reference purposes with no activity in this area during the 2004–5 reporting year.

TABLE VI-1

LISTING OF APPROVED UFTR STANDARD OPERATING PROCEDURES
(as of August 31, 2004)

0. ADMINISTRATIVE CONTROL PROCEDURES

0.1 Operating Document Controls (REV 3, 9/03)
0.2 Control of Maintenance (REV 5, 9/03)
0.3 Control and Documentation of UFTR Modifications (REV 1, 10/99)
0.4 10 CFR 50.59 Evaluation and Determination (REV 2, 7/00)
0.5 UFTR Quality Assurance Program (REV 3, 2/03)
0.6 Reactor Trip and Unscheduled Shutdown Review and Evaluation (REV 1, 4/02)
0.7 Control of NRC 10 CFR 50 Written Communications Requirements (REV 1, 12/97)
0.8 Operator Licensing Requalification Examination Controls (REV 2, 9/03)
0.9 Handling Incoming Suspicious Mail (Letters/Packages) and Shipments (REV 0, 3/04)

A. ROUTINE OPERATING PROCEDURES

A.1 Pre-Operational Checks (REV 16, 2/97)
A.2 Reactor Startup (REV 12, 5/87)
A.3 Reactor Operation at Power (REV 12, 11/94)
A.4 Reactor Shutdown (REV 11, 10/89)
A.5 Experiments (REV 4, 12/88)
A.6 Operation of Secondary Cooling Water (REV 4, 9/03)
A.7 Determination of Control Blade Integral or Differential Reactivity Worth (REV 2, 9/03)
A.8 Pneumatic Rapid Sample Transfer (Rabbit) System (REV 1, 10/99)

B. EMERGENCY PROCEDURES

B.1 Radiological Emergency (REV 5, 1/95)
B.2 Fire (REV 9, 1/95)
B.3 Threat to the Reactor Facility (Superseded by F-Series Procedures)
B.4 Flood (REV 2, 8/97)

C. FUEL HANDLING PROCEDURES

C.1 Irradiated Fuel Handling (REV 4, 2/85)
C.2 Fuel Loading (REV 5, 10/99)
C.3 Fuel Inventory Procedure (REV 4, 8/97)
C.4 Assembly and Disassembly of Irradiated Fuel Elements (REV 0, 9/84)

TABLE VI-1

LISTING OF APPROVED UFTR STANDARD OPERATING PROCEDURES
(as of August 31, 2004)

D. RADIATION CONTROL PROCEDURES

- D.1 UFTR Radiation Protection and Control (REV 5, 12/93)
- D.2 Radiation Work Permit (REV 11, 10/03)
- D.3 Primary Equipment Pit Entry (REV 4, 10/01)
- D.4 Removing Irradiated Samples from UFTR Experimental Ports (REV 7, 10/01)
- D.5 UFTR Reactor Waste Shipments: Preparations and Transfer (REV 2, 6/02)
- D.6 Control of UFTR Radioactive Material Transfers (REV 1, 4/00)
- D.7 Circulation, Sampling, Analysis, and Discharge of Holdup Tank Wastewater (REV 1, 4/02)

E. MAINTENANCE PROCEDURES

- E.1 Changing Primary Purification Demineralizer Resins (REV 5, 11/99)
- E.2 Alterations to Reactor Shielding and Graphite Configuration (REV 4, 4/02)
- E.3 Shield Tank and Shield Tank Recirculation System Maintenance (REV 2, 4/83)
- E.4 UFTR Nuclear Instrumentation Calibration Check (REV 3, 3/01)
- E.5 Superseded
- E.6 Argon-41 Concentration Measurement (REV 1, 10/03)
- E.7 Measurement of Temperature Coefficient of Reactivity (REV 1, 10/03)
- E.8 Verification of UFTR Negative Void Coefficient of Reactivity (REV 1, 4/02)

F. SECURITY PLAN RESPONSE PROCEDURES (Reactor Safeguards Material, Disposition Restricted)

- F.1 Physical Security Controls (Confidential, except for UFTR Form SOP-F.1A)
- F.2 Bomb Threat (Confidential, except for UFTR Form SOP-F.2A)
- F.3 Theft of (or Threat of the Theft of) Special Nuclear Material (Confidential, except for UFTR Form SOP-F.3A)
- F.4 Civil Disorder (Confidential)
- F.5 Fire or Explosion (Confidential)
- F.6 Industrial Sabotage (Confidential)
- F.7 Security Procedure Controls (REV 3, 4/02)
- F.8 UFTR Safeguards Reporting Requirements (REV 1, 12/97)
- F.9 Control of UFTR Vehicular Access (Confidential)

TABLE VI-2

LISTING OF APPROVED UFTR STANDARD OPERATING PROCEDURES
(as of August 31, 2005)

0. ADMINISTRATIVE CONTROL PROCEDURES

- 0.1 Operating Document Controls (REV 3, 9/03)
- 0.2 Control of Maintenance (REV 5, 9/03)
- 0.3 Control and Documentation of UFTR Modifications (REV 1, 10/99)
- 0.4 10 CFR 50.59 Evaluation and Determination (REV 2, 7/00)
- 0.5 UFTR Quality Assurance Program (REV 3, 2/03)
- 0.6 Reactor Trip and Unscheduled Shutdown Review and Evaluation (REV 1, 4/02)
- 0.7 Control of NRC 10 CFR 50 Written Communications Requirements (REV 1, 12/97)
- 0.8 Operator Licensing Requalification Examination Controls (REV 2, 9/03)
- 0.9 Handling Incoming Suspicious Mail (Letters/Packages) and Shipments (REV 0, 3/04)

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- A.3 Reactor Operation at Power (REV 12, 11/94)
- A.4 Reactor Shutdown (REV 11, 10/89)
- A.5 Experiments (REV 4, 12/88)
- A.6 Operation of Secondary Cooling Water (REV 4, 9/03)
- A.7 Determination of Control Blade Integral or Differential Reactivity Worth (REV 2, 9/03)
- A.8 Pneumatic Rapid Sample Transfer (Rabbit) System (REV 1, 10/99)

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- B.1 Radiological Emergency (REV 5, 1/95)
- B.2 Fire (REV 9, 1/95)
- B.3 Threat to the Reactor Facility (Superseded by F-Series Procedures)
- B.4 Flood (REV 2, 8/97)

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- C.1 Irradiated Fuel Handling (REV 4, 2/85)
- C.2 Fuel Loading (REV 5, 10/99)
- C.3 Fuel Inventory Procedure (REV 4, 8/97)
- C.4 Assembly and Disassembly of Irradiated Fuel Elements (REV 0, 9/84)

TABLE VI-2

LISTING OF APPROVED UFTR STANDARD OPERATING PROCEDURES
(as of August 31, 2005)

D. RADIATION CONTROL PROCEDURES

D.1 UFTR Radiation Protection and Control (REV 5, 12/93)
D.2 Radiation Work Permit (REV 11, 10/03)
D.3 Primary Equipment Pit Entry (REV 4, 10/01)
D.4 Removing Irradiated Samples from UFTR Experimental Ports (REV 7, 10/01)
D.5 UFTR Reactor Waste Shipments: Preparations and Transfer (REV 2, 6/02)
D.6 Control of UFTR Radioactive Material Transfers (REV 1, 4/00)
D.7 Circulation, Sampling, Analysis, and Discharge of Holdup Tank Wastewater (REV 1, 4/02)

E. MAINTENANCE PROCEDURES

E.1 Changing Primary Purification Demineralizer Resins (REV 5, 11/99)
E.2 Alterations to Reactor Shielding and Graphite Configuration (REV 4, 4/02)
E.3 Shield Tank and Shield Tank Recirculation System Maintenance (REV 2, 4/83)
E.4 UFTR Nuclear Instrumentation Calibration Check (REV 3, 3/01)
E.5 Superseded
E.6 Argon-41 Concentration Measurement (REV 1, 10/03)
E.7 Measurement of Temperature Coefficient of Reactivity (REV 1, 10/03)
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F.2 Bomb Threat (Confidential, except for UFTR Form SOP-F.2A)
F.3 Theft of (or Threat of the Theft of) Special Nuclear Material (Confidential, except for UFTR Form SOP-F.3A)
F.4 Civil Disorder (Confidential)
F.5 Fire or Explosion (Confidential)
F.6 Industrial Sabotage (Confidential)
F.7 Security Procedure Controls (REV 3, 4/02)
F.8 UFTR Safeguards Reporting Requirements (REV 1, 12/97)
F.9 Control of UFTR Vehicular Access (Confidential)
F.10 Background Investigations (Confidential)

VII. RADIOACTIVE RELEASES AND ENVIRONMENTAL SURVEILLANCE

This chapter summarizes the gaseous, liquid, and solid radioactive releases from the UFTR facility for this reporting year. Argon-41 is the primary gaseous release. Finally, this chapter includes a summary of personnel exposures at the UFTR facility.

A. Gaseous (Argon-41)

The gaseous releases from the UFTR facility for this reporting year are summarized in Table VII-1. The basis for the gaseous activity release values is indicated in Table VII-2. These values are obtained by periodic measurements of stack concentrations as required by Technical Specifications following UFTR SOP-E.6, "Argon-41 Concentration Measurements."

TABLE VII-1

UFTR GASEOUS RELEASE SUMMARY

Month	Release	Monthly Average Concentration
September 2004	3.1620 x 10^6 μCi/Month	1.1077 x 10^{-9} μCi/ml
October 2004	5.2758 x 10^6 μCi/Month	1.8483 x 10^{-9} μCi/ml
November 2004	10.2275 x 10^6 μCi/Month	3.5830 x 10^{-9} μCi/ml
December 2004	11.9530 x 10^6 μCi/Month	4.1875 x 10^{-9} μCi/ml
January 2005	8.5579 x 10^6 μCi/Month	2.9981 x 10^{-9} μCi/ml
February 2005	6.9251x 10^6 μCi/Month	2.3985 x 10^{-9} μCi/ml
March 2005	13.2625 x 10^6 μCi/Month	4.5935 x 10^{-9} μCi/ml
April 2005	5.0030 x 10^6 μCi/Month	1.7328 x 10^{-9} μCi/ml
May 2005	7.6512 x 10^6 μCi/Month	2.6500 x 10^{-9} μCi/ml
June 2005	5.4948 x 10^6 μCi/Month	1.9031 x 10^{-9} μCi/ml
July 2005	0.1769 x 10^6 μCi/Month	6.1254 x 10^{-11} μCi/ml
August 2005	0.0068 x 10^6 μCi/Month	2.3410 x 10^{-12} μCi/ml

TOTAL ARGON-41 Releases for the Reporting Year: 77.6965 Ci

YEARLY AVERAGE ARGON-41 Release Concentration: 2.2555 x 10^{-9} μCi/ml

UFTR Technical Specifications require the average Argon-41 release concentration averaged over a month to be less than 1.0 x 10^{-8} μCi/ml. All such monthly values are measured to be well below this limiting release concentration with an average monthly release concentration of 2.2555 x 10^{-9} μCi/ml. Even with the newest 10 CFR Part 20 values reducing the Argon-41 release concentration limit to 1.0 x 10^{-8} μCi/ml in January, 1994, there has been no problem expected as the highest monthly value (March 2005) listed in Table VII-1 is less than 46% of the allowable limit and the second highest (December 2004) is less than 42% of the allowable limit.

Total releases and average monthly concentrations are based upon periodic Argon-41 release concentration measurements made at equilibrium full power (100 kW) conditions. The results for these experimental measurements used in calculating the gaseous Argon-41 release data are summarized in Table VII-2. Entries in Table VII-2 represent the average results of analyses of a minimum of three (3) samples (usually four) per UFTR SOP-E.6 using a new gas standard obtained in response to NRC Inspection Report No. 88-01.

TABLE VII-2

UFTR GASEOUS RELEASE DATA TABLE

Month(s)	Releases per Unit Energy Generation	Instantaneous Argon-41 Concentration at Full Power [1]
Sep. 2004 - Jan. 2005	4087.00 μCi/kW-hr	10.309 x 10^{-8} μCi/ml
Feb. 2005- Aug. 2004	3736.34 μCi/kW-hr	9.3175 x 10^{-8} μCi/ml

Above limits ?

1E-8

[1]Values used to assure average release concentration meets 10 CFR 20 limits.

B. Liquid Waste from the UFTR/Nuclear Sciences Complex

The UFTR normally releases about one (1) liter of primary coolant per week to the holdup tank as waste from primary coolant sampling. A total of 52 weekly samples were taken during this reporting year; the average activity for these coolant samples was 2.48 x 10^{-7} μCi/ml (β-γ) and 6.45 x 10^{-9} μCi/ml (α) for this 2004-2005 reporting period. There were two discharges from the Wastewater Holdup Tank for this reporting period. On October 15, 2004, a total of 3363 liters was discharged. The discharge contained less than 1.00 x 10^{-3} μCi of Total Activity, less than 1.00 x 10^{-3} μCi of Dissolved Activity, and less than 1.00 x 10^{-3} μCi Activity of Suspended Solids all of which were less than the Lower Limit of Detection. On June 10, 2005, a total of 3290 liters were discharged. The discharge contained less than 1.00 x 10^{-3} μCi of Total Activity, less than 1.00 x 10^{-3} μCi of Dissolved Activity, and less than 1.00 x 10^{-3} μCi Activity of Suspended Solids which was also less than the Lower Limit of Detection.

C. Solid Waste Shipped Off-site

The UFTR facility made no shipments of solid waste during this 2004–5 reporting year. The last two shipments of solid waste from the UFTR were made on December 10, 1985 and June 20, 2002.

The shipment of solid waste that was made on December 10, 1985 was through ADCO Services, Inc. and consisted of one 55-gallon drum containing radioactive scrap metal parts as well as paper, plastic, and other reactor-related waste materials associated primarily with the work to restore proper functioning of the UFTR control blade drive systems. The activity of the shipment was approximately 3.125 Curies with the activity primarily attributed to Cobalt-60.

Though a similar shipment of two drums had been planned for about fifteen reporting years to remove all of the products resulting from the control blade restoration and maintenance project of 1985–86, this shipment had not occurred prior to the 2001–2 reporting year. With waste consolidated for shipment to clear space for waste expected to be generated during the UFTR conversion from HEU to LEU fuel expected within the next five years, the new Standard Operating Procedure UFTR SOP-D.5, "UFTR Reactor Waste Shipments: Preparations and Transfer" originally generated in the 1986–87 reporting year and revised in April, 1992 was updated and used along with guidance provided in several NRC Information Notices published in the previous several years to assure proper control of the waste shipment. Therefore, for the 2001–2 reporting year, the UFTR facility shipped fourteen 55-gallon drums containing radioactive scrap metal parts, paper, plastic, protective clothing, and other reactor-related waste materials on June 20, 2002. Table VII-3 gives the total activity for each of the fourteen drums that were shipped out to the centralized radioactive waste handling facility on the University of Florida Campus.

No waste has been shipped from the reactor license since the 2001–2 reporting year.

TABLE VII-3

RADIOACTIVE REACTOR WASTE

Container	Cobalt -60 Total Activity (μ Ci)	Silver-110 Total Activity (μ Ci)
1	18.7	
2	499.9	1.7
3	12.2	
4	28.8	
5	9.9	
6	6.6	
7	13.6	
8	9.4	
9	6.2	
10	17.3	
11	19.9	
12	12.8	
13	12.5	
14	7.4	

D. Environmental Monitoring

The UFTR maintains continuous Luxel dosimeter monitoring in areas adjacent to and in the vicinity of the UFTR complex. The cumulative totals for this reporting year from September 2004 to August 2005 along with months for non-zero values are summarized in Table VII-4A. Overall, the values in Tables VII-4A and VII-4B show minimal environmental radiation dose from UFTR operations. The recorded TLD exposures are essentially background to within the accuracy of the monitoring instruments.

The accumulation of exposure recorded by month of exposure on the monitoring badges is presented in Table VII-4B. The values recorded in Tables VII-4A and VII-4B are considered to support the conclusion of minimal environmental exposures from UFTR operations.

TABLE VII-4A

CUMULATIVE RESULTS OF ENVIRONMENTAL MONITORING
SEPTEMBER 1, 2004 TO AUGUST 31, 2005

Luxel Dosimeter Designation	Total Exposure (mrem) [1]	Month(s) of Exposure
1	8	10/04, 11/04, 2/05, 7/05, 8/05
2	4	1/05, 7/05, 8/05
3	M	--
4	M	--
5	4	1/05, 7/05
6	M	--
7	1	7/05
8	M	--
9	M	--
10	M	--
11	M	--
12	M	--
13	M	--

[1]M denotes minimal (<1 mrem) exposure.

TABLE VII-4B

LUXEL DOSIMETER
EXPOSURE RECORD BY MONTH OF EXPOSURE [1,2]

TLD Number	Sep 04 (mrem)	Oct 04 (mrem)	Nov 04 (mrem)	Dec 04 (mrem)	Jan 05 (mrem)	Feb 05 (mrem)	Mar 05 (mrem)	Apr 05 (mrem)	May 05 (mrem)	Jun 05 (mrem)	Jul 05 (mrem)	Aug 05 (mrem)
1	M	1	2	M	M	1	M	M	M	M	2	2
2	M	M	M	M	2	M	M	M	M	M	1	1
3	M	M	M	M	M	M	M	M	M	M	M	M
4	M	M	M	M	M	M	M	M	M	M	M	M
5	M	M	M	M	2	M	M	M	M	M	2	M
6	M	M	M	M	M	M	M	M	M	M	M	M
7	M	M	M	M	M	M	M	M	M	M	1	M
8	M	M	M	M	M	M	N/A	M	M	M	M	M
9	M	M	M	M	M	M	M	M	M	M	M	M
10	M	M	M	M	M	M	M	M	M	M	M	M
11	M	M	M	M	M	M	M	M	M	M	M	M
12	M	M	M	M	M	M	M	M	M	M	M	M
13	M	M	M	M	M	M	M	M	M	M	M	M

[1]M denotes minimal (<1 mrem) exposure.
[2]Entrys marked N/A refers to lost luxel dosimeter.

E. Personal Radiation Exposure

UFTR-associated personnel exposures greater than minimum detectable during the reporting period are summarized in this section.

Table VII-5 lists the permanent whole-body badge exposures recorded above background for the reporting year for personnel employed directly at the UFTR. These exposures are summarized for all badged personnel on an annual basis.

TABLE VII-5

ANNUAL UFTR PERSONNEL EXPOSURE

Name	Position	Permanent Badge Exposure (mrem) [1,2]
W. Vernetson	Facility Director/Senior Reactor Operator	2
B. Shea	Senior Reactor Operator (9/04 – 4/05)	7
M. Berglund	Reactor Operator Trainee	M
R. Leug	Reactor Operator Trainee (9/04 – 4/05)	5

[1] The exposure recorded here is for deep/whole-body dose.
[2] M denotes minimal (<1 mrem) exposure.

Table VII-6 lists the permanent whole-body badge exposures recorded above background for the reporting year for non-permanent personnel employed at the UFTR. These exposures are summarized for all badged non-permanent UFTR personnel on an annual basis with no further breakdown because all exposures are well below 100 mrem for the year and in most cases are minimal.

TABLE VII-6

ANNUAL NON-PERMANENT UFTR PERSONNEL EXPOSURE

Name	Position [1]	Permanent Badge Exposure (mrem) [2,3]
M. Crawford	NAA Lab/ Reactor Facility Technician (9/04 – 5/05)	M
M. Holman	NAA Lab/Reactor Facility Technician	6
G. Joseph	NAA Lab/Reactor Facility Technician	4
N. Patel	NAA Lab/ Reactor Facility Technician (1/05 – 8/05)	M
M. Yenatskyy	NAA Lab/ Reactor Facility Technician (5/05 – 8/05)	M
B. Stewart	NAA Lab/ Reactor Facility Technician (7/05 – 8/05)	M

[1]All are part time employed students.
[2]The exposure recorded here is for deep/whole-body dose.
[3]M denotes minimal (<1 mrem) exposure.

Table VII-7 lists the prompt reading dosimeter exposures recorded for visitors, students, or other non UFTR personnel. Few individuals had greater than 1 mrem prompt reading dosimeter exposure measurement over the entire reporting period as indicated in Table VII-7.

TABLE VII-7

EXPOSURE RECORDS FOR UFTR VISITORS
AS RECORDED BY PROMPT-READING DOSIMETERS

Personnel[1]	Date	Exposure (mrem) [1]	Comments
S. Turner	9/29/04, 5/01/05	2	Experimenter (Industry)
T. Palmer	10/21/04, 10/29/04	3	Experimenter (Student)

[1]All exposure readings are for whole-body exposures recorded > 1 mrem.

It should be noted that tours of reactor facilities are strictly controlled and limited during periods when the reactor is running or ports are open or other opportunities for significant radiation fields are present. Therefore, the lack of visitor exposure is expected and in agreement with ALARA guidelines.

APPENDIX A

UFTR TECHNICAL SPECIFICATION AMENDMENT #24
NRC APPROVAL PACKAGE
INCLUDING COVER LETTER FROM NRC,
APPROVED LICENSE AMENDMENT,
AND NRC SAFETY EVALUATION

ATTACHMENT II

RECEIVED JAN 1 0 2005

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

January 4, 2005

Dr. William G. Vernetson
Director of Nuclear Facilities
University of Florida
Department of Nuclear and
  Radiological Engineering
202 Nuclear Sciences Center
P.O. Box 118300
Gainesville, FL 32611-8300

SUBJECT: UNIVERSITY OF FLORIDA—AMENDMENT RE: FUEL AND CONTROL BLADE SURVEILLANCE (TAC NO. MC4483)

Dear Dr. Vernetson:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 24 to Amended Facility Operating License No. R-56 for the University of Florida Training Reactor. The amendment consists of changes to the technical specifications (TSs) in response to your application of September 17, 2004, as supplemented on December 15, 2004.

The amendment increases the interval of the surveillance for the control blades and drive system and the in-core reactor fuel elements.

A copy of the safety evaluation supporting Amendment No. 24 is also enclosed.

Sincerely,

Alexander Adams

Alexander Adams, Jr., Senior Project Manager
Research and Test Reactors Section
New, Research and Test Reactors Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 50-83

Enclosures: 1. Amendment No. 24
            2. Safety Evaluation

cc w/enclosures: See next page

cc:

Dr. Alireza Haghighat, Chairman
Nuclear & Radiological Engineering Department
University of Florida
202 Nuclear Sciences Center
P.O. Box 118300
Gainesville, FL 32601-8300

Administrator
Department of Environmental Regulation
Power Plant Siting Section
State of Florida
2600 Blair Stone Road
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State Planning and Development
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Office of Planning and Budgeting
Executive Office of the Governor
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William G. Vernetson, Ph.D.
Director of Nuclear Facilities
University of Florida
202 NSC/PO Box 118300
Gainesville, FL 32611-8300

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

UNIVERSITY OF FLORIDA

DOCKET NO. 50-83

AMENDMENT TO AMENDED FACILITY OPERATING LICENSE

Amendment No. 24
License No. R-56

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

   A. The application for an amendment to Amended Facility Operating License No. R-56 filed by the University of Florida (the licensee) on September 17, 2004, as supplemented on December 15, 2004, conforms to the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the regulations of the Commission as stated in Chapter I of Title 10 of the *Code of Federal Regulations* (10 CFR);

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

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

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

   E. This amendment is issued in accordance with the regulations of the Commission as stated in 10 CFR Part 51, and all applicable requirements have been satisfied; and

   F. Prior notice of this amendment was not required by 10 CFR 2.105 and publication of a notice for this amendment is not required by 10 CFR 2.106.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment, and paragraph 2.C.(2) of Amended Facility Operating License No. R-56 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 24, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Patrick M. Madden

Patrick M. Madden, Chief
Research and Test Reactors Section
New, Research and Test Reactors Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Enclosure: Appendix A, Technical
Specifications Changes

Date of Issuance: January 4, 2005

ENCLOSURE TO LICENSE AMENDMENT NO. 24

AMENDED FACILITY OPERATING LICENSE NO. R-56

DOCKET NO. 50-83

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

Remove	Insert
19	19
21	21

Table 4.1 Control blade withdrawal inhibit interlocks operability tests

Inhibit	Limit	Frequency
Reactor period	≤ 10 sec	Daily checkout
Safety channels and wide range drawer not in OPERATE position	-	Daily checkout
Multiple blade withdrawal	Any 2 or more blades simultaneously in Manual	Daily checkout
	Any 2 safety blades In Automatic	Daily checkout
Source count rate	<2 cps	Verification only when count rate <2 cps during daily checkout

(4) The mechanical integrity of the control blades and drive system shall be inspected during each incore inspection but shall be fully checked at least once every 10 years at intervals not to exceed 12 years.

(5) Following maintenance or modification to the control blade system, an operability test and calibration of the affected portion of the system, including verification of control blade drive speed, shall be performed before the system is to be considered operable.

(6) The reactor shall not be started unless (a) the weekly checkout has been satisfactorily completed within 7 days prior to startup, (b) a daily checkout is satisfactorily completed within 8 hr prior to startup, <u>and</u> (c) no known condition exists that would prevent successful completion of a weekly or daily check.

(7) The limitations established under Paragraph 4.2.2(6)(a) and (b) can be deleted if a reactor startup is made within 6 hr of a normal reactor shutdown on any one calendar day.

(8) The following channels shall be calibrated annually, at intervals not to exceed 13 months, and any time a significant change in channel performance is noted:

(a) log N - period channel
(b) power level safety channels (2)
(c) Linear power level channel

4.2.6 Reactor Building Evacuation Alarm Surveillance

(1) The coincidence automatic actuation of two area monitors and the manual actuation of the evacuation alarm shall be tested as part of the weekly checkout.

(2) The automatic shutoff of the air conditioning system and the reactor vent system shall be tested as part of the weekly checkout.

(3) Evacuation drills for facility personnel shall be conducted quarterly, at intervals not to exceed 4 months, to ensure that facility personnel are familiar with the emergency plan.

4.2.7 Surveillance Pertaining to Fuel

(1) The incore reactor fuel elements shall be inspected every 10 years at intervals not to exceed 12 years, in a randomly chosen pattern, as deemed necessary. At least 8 elements will be inspected.

(2) Fuel-handling tools and procedures shall be reviewed for adequacy before fuel loading operations. The assignment of responsibilities and training of the fuel-handling crew shall be performed according to written procedures.

4.2.8 Primary and Secondary Water Quality Surveillance

(1) The primary water resistivity shall be determined as follows:

(a) Primary water resistivity shall be measured during the weekly checkout by a portable Solu Bridge using approved procedures. The measured value shall be larger than 0.4 megohm-cm.

(b) Primary water resistivity shall be measured during the daily checkout at both the inlet and outlet of the demineralizers (DM). The measured value, determined by an online Solu Bridge alarming in the control room, shall be larger than 0.5 megohm-cm at the outlet of the DM.

(2) The primary water radioactivity shall be measured during the weekly checkout for gross β-γ and gross α activity.

(a) The measured α activity shall not exceed 50 dpm above background level.

(b) The measured β-γ activity shall not exceed 25% above mean normal activity level.

(3) The secondary water system shall be tested for radioactive contamination during the weekly checkout according to written procedures.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 24 TO

AMENDED FACILITY OPERATING LICENSE NO. R-56

THE UNIVERSITY OF FLORIDA

DOCKET NO. 50-83

1.0 INTRODUCTION

By letter dated September 17, 2004, as supplemented on December 15, 2004, the University of Florida (UF or the licensee) submitted a request for amendment to Amended Facility Operating License No. R-56 for the UF Training Reactor. The request would change the Technical Specification (TS) interval of fuel element inspection from at least 4 elements every 5 years with the interval between inspections not to exceed 6 years to at least 8 elements every 10 years with the interval between inspections not to exceed 12 years. The request would also change the TS interval of inspection and checks of the control blades and drive system from every 5 years with the interval between inspections not to exceed 6 years to every 10 years with the interval between inspections not to exceed 12 years.

2.0 BACKGROUND

The UF operates an Argonaut-type research reactor with a maximum licensed power level of 100 kW(t). The reactor uses Material Testing Reactor (MTR)-type plate fuel. A design feature of the Argonaut-type research reactor is a massive biological shield over the reactor core that makes access to the core area of the reactor very difficult. On December 28, 2001, Amendment No. 23 to the UF license was issued which changed the same TSs surveillance requirements that the licensee is requesting be changed by the current amendment request. Amendment No. 23 increased the interval of fuel element inspection and control blades and drive system inspection and checks from biennially to the current five years.

3.0 EVALUATION

The regulations in 10 CFR 50.36 require nuclear reactors to have TSs. The regulations require TSs to include surveillance requirements which are discussed in 10 CFR 50.36(c)(3). The regulation states that surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. The staff has determined that the changes proposed by the licensee continue to meet the requirements of 10 CFR 50.36.

The UF has requested a change to TS 4.2.7 (1) for surveillance of in-core reactor fuel elements that would change the interval on the surveillance from every 5 years (interval not to exceed 6 years) to every 10 years (interval not to exceed 12 years) and would increase the minimum number of fuel elements inspected per inspection from 4 to 8. TS 4.2.7 (1) currently reads:

> The incore reactor fuel elements shall be inspected every 5 years at intervals not to exceed 6 years, in a randomly chosen pattern, as deemed necessary. At least 4 elements will be inspected.

The licensee has proposed changing this TS to read:

> The incore reactor fuel elements shall be inspected every 10 years at intervals not to exceed 12 years, in a randomly chosen pattern, as deemed necessary. At least 8 elements will be inspected.

TS 4.2.2(4) concerning surveillance of the control blades and drive system reads as follows:

> The mechanical integrity of the control blades and drive system shall be inspected during each incore inspection but shall be fully checked at least once every 5 years at intervals not to exceed 6 years.

The licensee has proposed changing this TS to read:

> The mechanical integrity of the control blades and drive system shall be inspected during each incore inspection but shall be fully checked at least once every 10 years at intervals not to exceed 12 years.

The licensee has requested this change to reduce the considerable effort needed to carry out the inspection. Access to the core of an Argonaut reactor requires disassembly of the primary shielding which consists of a number of large, heavy shield blocks. The fuel inspection process takes about two weeks to accomplish. This change to TS 4.2.7 (1) also impacts TS 4.2.2 (4). Technical Specification 4.2.2 (4) has requirements to perform control blades and drive system inspections and checks. The checks have an interval stated in the TS (currently 5 years with the licensee proposing changing to 10 years) but the inspection interval is tied to the performance of in-core inspections. The proposed change to the fuel element inspection interval would also change the interval for control blades and drive system inspections from 5 to 10 years. The licensee would carry out both inspections and checks on the control blades and drive system during the same reactor disassembly for fuel inspection.

The purpose of the fuel surveillance is to reduce the possibility of operating the reactor with failed fuel. The surveillance consists of a visual inspection of the fuel elements. This inspection would only detect gross problems with fuel elements and would not detect pin hole defects in the fuel, the most likely fission product release path. Over 33 years of fuel element inspections have not found any failed fuel. The primary indication of cladding failure is the presence of fission products in the primary coolant. TS 4.2.8 (2) requires weekly measurement of primary water radioactivity. TS 3.7 (4) prohibits reactor operation if there is evidence of fuel element failure and TS 3.7 (3) requires fuel elements exhibiting the release of fission products to be removed from the core. TS 3.7 (3) states that fission product contamination of the primary water shall be treated as evidence of fuel element failure. Indication of fission products in the primary water would require disassembly of the reactor to locate and remove the failed fuel element. The staff's experience with fuel failures in research reactors had been that the overwhelming majority of failures have been detected by monitoring of the primary coolant for fission products.

The surveillance interval proposed by the licensee would not result in any decrease of total fuel elements inspected over time. For example, for a 20-year period, both the current and proposed surveillance interval would require inspection of 16 elements.

The licensee discusses several advantages to the proposed TS: reduction in wear and tear on the reactor from reduced disassembly, reduced radiation dose to the facility staff, and increased efficiency of operation and utilization.

The inspection of the control blades and drive system consists of a visual inspection of the in-core components of the system. The check of the system consists of the visual inspection of the in-core components and partial disassembly of drive system components such as gearboxes to check for oil level, hardened grease, foreign matter and wear. Control blade drop times, controlled insertion times and withdrawal times are measured when the reactor is reassembled to help ensure proper reassembly. The visual inspections of the in-core control blades and drive system have never identified any problems. In addition, the regular measurement of control blade drop times (semiannually), controlled insertion times (semiannually) and withdrawal times (weekly) would provide indication of system degradation between inspections and checks of the system. The advantages listed above for an increased surveillance interval for the fuel elements also apply to the control blades and drive system inspections and checks.

The NRC staff concludes that the existing requirement on primary coolant monitoring for fission products will detect failed fuel in an acceptable manner. Based on these existing requirements, the fact that the licensee will continue to inspect fuel elements, a reduction in radiation dose to the reactor staff, and the reduction of wear and tear on the reactor from reduced disassembly, the proposed change in the surveillance requirement for fuel element inspection is acceptable to the staff. Also based on the history of positive results of visual inspections of the control blades and drive system and the existing surveillance on control blade drop times, controlled insertion times and withdrawal times, the staff concludes that degradation of the control blades and drive systems will be detected in an acceptable manner and that the interval for inspection and check of the control blades and drive system can be increased from 5 to 10 years.

The proposed changes to the TSs also increase the "not to exceed" interval to 12 years. This gives the licensee flexibility in performance of the surveillance. ANS/ANSI 15.1-1990, "American National Standard for the Development of Technical Specifications for Research Reactors," which is supported by the NRC staff for research and test reactor TS format contains suggested "not to exceed" intervals. While a 10-year surveillance interval is not addressed in the standard, the "not to exceed" interval for a 5-year surveillance is 6 years. The licensee has proposed a 12-year "not to exceed" interval for the proposed 10-year surveillance. This is double the time for a 5-year surveillance and is consistent with the standard. Because the proposed surveillance interval is consistent with that given in ANS/ANSI 15.1-1990, the addition of a maximum 12-year surveillance interval is acceptable to the staff.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in inspection and surveillance requirements. The staff determines that this amendment involves no significant hazards consideration, no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off site, and no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

The staff concludes, on the basis of the considerations discussed above, that (1) the amendment does not involve a significant hazards consideration because the amendment does not involve a significant increase in the probability or consequences of accidents previously evaluated, create the possibility of a new kind of accident or a different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed activities; and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

Principal Contributor: A. Adams, Jr.

Date: January 3, 2005

APPENDIX B

NRC REQUEST FOR ADDITIONAL INFORMATION PACKAGE
FOR UFTR RELICENSING SUBMITTAL
INCLUDING LETTER FROM NRC
WITH ENCLOSED LISTING OF 92 QUESTIONS

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

ATTACHMENT IV

April 6, 2005

RECEIVED APR 1 1 2005

Dr. William G. Vernetson
Director of Nuclear Facilities
University of Florida
Department of Nuclear and
Radiological Engineering
202 Nuclear Sciences Center
P.O. Box 118300
Gainesville, FL 32611-8300

SUBJECT: UNIVERSITY OF FLORIDA—REQUEST FOR ADDITIONAL INFORMATION RE: LICENSE RENEWAL FOR THE UNIVERSITY OF FLORIDA TRAINING REACTOR (TAC NO. MB 5804)

Dear Dr. Vernetson:

We are continuing our review of your request for renewal of Amended Facility License No. R-130 for the University of Florida Training Reactor which you submitted on July 29, 2002. During our review of your request, questions have arisen for which we require additional information and clarification. Because of the recent Department of Energy decision to move forward on conversion of the University of Florida Training Reactor from high enriched to low enriched uranium fuel, the staff will discuss with you the timing of your responses to these questions. In accordance with 10 CFR 50.30(b), your response must be executed in a signed original under oath or affirmation. Following receipt of the additional information, we will continue our evaluation of your amendment request.

If you have any questions regarding this review, please contact me at (301) 415-1127.

Sincerely,

Alexander Adams

Alexander Adams, Jr., Senior Project Manager
Research and Test Reactors Section
New, Research and Test Reactors Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 50-83

Enclosure: As stated

cc w/enclosure: See next page

cc:

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REQUEST FOR ADDITIONAL INFORMATION
UNIVERSITY OF FLORIDA TRAINING REACTOR
DOCKET NO. 50-83

1 THE FACILITY

1-1 The Introduction to the SAR does not discuss shared facilities and equipment. Does this mean that the University of Florida Training Reactor (UFTR) does not have shared facilities and equipment as discussed in Section 1.4 of NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors?" Please confirm that there are no shared facilities and equipment or describe the shared facilities and equipment.

2 SITE CHARACTERISTICS

2-1 Section 2.1.1.2, Boundary and Zone Area Maps, page 2-1, and Section 2.1.1.3, Boundaries for Establishing Effluent Release Limits, page 2-2. Reference is made to definitions from 10 CFR Part 100. This regulation is not applicable to research reactors. How does the facility and site meet the definitions in 10 CFR Part 20 (e.g., restricted area and site boundary) and 10 CFR Part 73 (e.g., protected area)? The area of the facility and site proposed under the reactor license should be clearly described.

2-2 Is there any railroad station or line located near the UFTR site so that a derailment accident could affect the reactor building?

2-3 Is there any military installation (e.g., aircraft flight path) near the UFTR site so that military activities in the area could affect the reactor building?

2-4 Local meteorological measurements for use in evaluating accidental effluent releases from the UFTR do not appear to be available. Explain where this information will be obtained if needed.

3 DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS

3-1 Section 3.1, Design Criteria, page 3-1. The UFTR reactor building is divided into two distinct areas. The reactor area is 30 ft. by 60 ft. by 29 ft. high and is located at the north end of the building. The remaining area of the building is used for research and teaching laboratories, faculty and graduate student offices, and work areas. The reactor area is on a one foot thick slab resting on undisturbed or compacted earth. The thickness of this slab is increased to 18 inches under the reactor. The walls of the reactor area are constructed of one foot thick monolithic reinforced concrete resting on mat footings. The 3-inches thick roof of the reactor area is built-up of a precast roof tile supported by steel-bar joists spaced 2 ft. on centers. Please discuss the following:

a. What building Code was used while constructing the reactor building? Did the building design include any seismic and wind loads?

b. What administrative controls exist regarding the use of the overhead crane during reactor operations?

c. Describe the design of the brick flue and/or the reactor stack that carries the exhaust air above the top of the reactor building. How tall is the stack?

4 REACTOR DESCRIPTION

4-1 Section 4.1.1, General Reactor System Design. Demineralized water is used as the primary coolant. Has the UFTR experienced any water chemistry excursions which have resulted in material degradation of the fuel or other core components?

4-2 Section 4.1.2, Design and Performance Characteristics. The SAR addresses nuclear and thermal design characteristics. Discuss if other issues may limit fuel integrity, including water chemistry issues, physical stresses from mechanical or hydraulic forces, fuel burnup, radiation damage to fuel, and fission product retention.

4-3 Section 4.1.2, Design and Performance Characteristics. The transmittal letter (Dr. W. G. Vernetson to NRC, dated July 25, 2002), states that the reason for the change on control blade drop time from 1.0 seconds to 1.5 seconds was to prevent unnecessary unstacking and entry into the core to make repairs to assure meeting the 1.0 second limit.

a. Why were the control blades not able to meet the 1.0 second limit?

b. What is the quantitative impact on the reactor safety margins due to this change?

c. Section 4.2.2.1, Table 4-1: Table 4-1states that the worth of the three safety shim arms are as follows: #1, 122(sic)% $\Delta k/k$; #2, 1.35% $\Delta k/k$; and #3, 1.83% $\Delta k/k$. However in the technical specifications (TSs) (Section 5.5) the blade worths are stated to be between about 1.3 and 2.3% $\Delta k/k$. Please make the SAR and TSs consistent.

4-4 Section 4.2.1, Fuel System Design. The TSs (Appendix 14.1, Section 5.3.1) permit several fuel matrix fabrication options. Which of these processes was used for the UFTR fuel? Do the different fuel fabrication options present any unique issues or limitations regarding the use of the fuel for the UFTR?

4-5 Sections 4.2.1, Fuel System Design and 4.2.2.1, Control Rods. What is the lifetime of the fuel assembly and control rods at UFTR? How does the control rod worth change with time?

4-6 Section 4.2.2.1, Control Rods. This section states the usual detailed information is not provided since the 'control rod systems are previously operated systems', however no information on this system is discussed in Section 16.1, Prior Use of Reactor Components. Please provide a reference which discusses the operating history of the control rods.

4-7 Section 4.2.3, Neutron Moderator and Reflector. The SAR alludes to aging effects and the hope to load new reactor grade graphite into the UFTR core. Discuss the aging effects observed, and any operating changes or restrictions which have been needed in response to the aging effects. What is the remaining life of the existing graphite?

4-8 Section 4.2.4, Neutron Startup Source. What special handling restrictions are in place applicable to the neutron startup sources? Where are the sources stored when not in use?

4-9 Section 4.2.5, Core Support Structure. The SAR states that the core support structure materials will continue to be adequate given the current operating conditions. What is the estimated remaining life for the core support structure?

4-10 Section 4.3, Reactor Tank. How would leakage from the aluminum reactor tanks be detected? What is the minium leakage rate that can be detected, and what is the maximum time duration that this leakage can occur before detection? What would the impact on public health and safety be from tank leakage?

4-11 Section 4.3, Reactor Tank. This section discusses a fuel safety limit of 200°F. It appears that the purpose of the limit is to protect the structural integrity of the reactor tank and not the integrity of the fuel cladding. Please clarify.

4-12 Section 4.4 Biological Shielding. Is the addition of Poly-B-Pb in the shielding going to happen? If it is, its addition should be discussed in more detail.

4-13 Section 4.4, Biological Shielding, Page 4-10 states the actual exposure at the north and south faces is approximately 3mR/hr, but Table 4-4 shows 2 and 0.8 mR/hr at 1 foot. Is the 3mR/hr on contact and the table value measured at a distance from the reactor?

4-14 Section 4.4, Biological Shielding. Is ground water and soil activation possible? If so, please discuss.

4-15 Section 4.4. Is 2.5 mR/hr the normal and expected dose rate at the three area monitors or is that an unusual level?

4-16 Section 4.4, Biological Shielding. Could radiation damage and heating of the shielding during the 20-year renewal period along with potential radiation-induced degradation and activation of the material impact the integrity of the shielding? Is there any potential for streaming of radiation along the shielding? Discuss shielding of experimental facilities, if any.

4-17 Section 4.4, Biological Shielding. Please address the shielding of spent fuel.

4-18 Section 4.5.1, Normal Operating Conditions. What administrative (operating procedures and TS limits) and physical constraints (interlocks and trips) exist to prevent inadvertent addition of positive reactivity?

4-19 Section 4.5.1.1, Flux Distribution. There seems to be $\sim 10^{12}$ difference in the fluxes quoted in Table 4-6 and Figures 4-23 and 4-24. Please address.

4-20 Section 4.5.1.1, Flux Distribution. Please state the neutron energy divisions for the 4 group calculations.

4-21 Section 4.5.1.2, Control Blade Worth, Shutdown Marging and Excess Reactivity. The values for the control rod worths are not consistent between Tables 4-1 and 4-9. Please explain the difference.

4-22 Section 4.5.2, Reactor Core Physics Parameters. For what type of core were these coefficients determined, fresh, end-of-life or, some other condition?

4-23 Section 4.5.2, Reactor Core Physics Parameters. Do the parameters change significantly with burn up?

4-24 Section 4.5.2, Reactor Core Physics Parameters. Please describe the axial and radial flux densities.

4-25 Section 4.5.3, Operating Limits. This section of the SAR should contain the discussion and calculations to support the safety limits, limiting safety systems settings, limiting conditions of operation and surveillance requirements related to operation. (See Pages 4-11 and 4-12 of NUREG-1537, Part 1, for a list of detailed information that is expected to be included in this section covering Operating Limits.) Please provide this information or provide references where this information can be found.

5 REACTOR COOLANT SYSTEMS

5-1 Section 5.2, Primary Coolant System, page 5-1. The UFTR is designed for forced flow cooling while in operation. There is a heat exchanger (HX) in the forced flow loop of the primary coolant system to maintain the primary coolant temperature. The primary coolant cleanup system loop is also part of the primary coolant system. The cleanup pump in this loop is interlocked with the primary pump to prevent its operation during normal operation of the system. The function of this cleanup system is to maintain the chemistry quality and conductivity of the primary coolant. The heat exchanger is cooled by an open loop secondary cooling system which uses deep well water to cool the primary coolant and discharges into the city storm sewer system. Please discuss the following:

a. Provide sketches or layout drawings to depict the location of the primary coolant system and associated systems (i.e., secondary coolant, primary coolant cleanup and primary coolant makeup water systems) with respect to the building structures of the reactor building. Specifically, identify the portions of these systems including associated major components that are located inside and outside of the reactor building confinement.

b. If there were a reactor coolant piping/component failure outside of the reactor core, describe where the primary water would be collected in the building? How would this be detected, measured, and alarmed?

c. It is stated that the graphite rupture disk is set to burst at 7 psia, which is 2 psi above normal operating pressure. What is the normal operating pressure of the primary coolant? Note that page 3-5 states that the system operates at ambient pressure and a low temperature below 155°F.

5-2 Section 5.3, Secondary Coolant System, page 5-3. The pressure of the secondary water is maintained higher than the primary system to prevent contamination of secondary coolant. What are the normal operating pressures of the primary and secondary coolant systems? How are these pressures monitored? If a leak were to develop in the primary/secondary boundary, how would this be detected? Since the secondary water is tested weekly for radiological contamination (Appendix 14.1, Section

4.3, Item (4)), is there any way to identify such contamination that may be occurring in-between the weekly testing period (e.g., on continuous basis)?

5-3 Section 5.3, Secondary Coolant System. Normal secondary flow is 200 gpm. At 140 gpm a low flow warning signal is sent to the control room and at 60 gpm a reactor trip is initiated if the reactor is at or above 1 kW after a 10-sec warning. When city water is used, a less than 8 gpm flow in the input line will initiate a reactor trip for power levels above 1 kW. What is the normal flow rate of city water when it is used as the backup to well water?

5-4 Section 5.6, Nitrogen-16 Control System, page 5-4. Is there any shielding around the piping from the reactor to the coolant storage tank and the coolant storage tank area? If not, explain any administrative procedures to restrict entry into this area during reactor operation and to allow time for N-16 decay?

6 ENGINEERED SAFETY FEATURES

6-1 Section 6 states that because the reactor is self-limiting, there is no additional requirement for engineered safety features. Confinement is achieved through keeping a negative pressure on the control and reactor rooms. Dilution is used to keep both postulated accident and operational radioactivity releases within specifications. TS 3.4, Reactor Vent System, states that this system shall be operational during operation of the reactor. What is the purpose of this system? Since the backup to control blade insertion is allowing the water (the moderator) to run out of the fuel boxes making the reactor sub-critical, is the vent system heat removal capability required to prevent cladding damage to the fuel? If this system was credited as a heat removal mechanism for accident mitigation it should be considered an ESF. Please explain if this is the case, it is not clear in the SAR.

7 INSTRUMENTATION & CONTROL

7-1 Section 7.1, fourth paragraph, states that "...system instruments are hardwired analog instrument type with the exception of the temperature monitor and record system which is a digital system instrument type." Section 7.2.1 indicates that the control blade position indicators and master console clock are now also digital instruments/displays. Please clarify.

7-2 Sections 7.2.3.4.2 , 7.3.2, and 5.3 list a low secondary coolant system flow trip of 60 gpm when the deep well pump is the coolant water source and the reactor is operating above 1 kW. The low flow trip first illuminates a red scram warning light on the reactor control console and then trips the reactor after approximately a 10 second delay. What is the basis of the 10 second delay?

7-3 Sections 7.2.3.4.2, 7.3.2, and 5.3 list a low secondary coolant system flow trip of 8 gpm when the city water supply is the coolant water source and the reactor is operating above 1 kW. Please explain the difference in the low flow setpoints when using the two different sources of secondary cooling water. Also, the last sentence in paragraph four of Section 5.3 is not clear: Is there a time delay associated with the city water 8 gpm low flow trip? If so, what is the basis of this time delay?

7-4 Figure 7-10 shows the temperature monitor and recorder system. Are any of the reactor scram or alarm functions dependent on software? If so, what validation and verification process was used on the software. It appears that a CPU is used in the monitor temperature virtual instrument. This system appears to be digital based. If so, what validation and verification process was used on the software. Does the reactor operator make operational decisions based on the output of the monitor temperature virtual instrument? Does the instrument store temperature data that is used to show compliance with license requirements?

7-5 Section 7.2.3.4.2 states that there is a key operated switch inside the reactor control console rear door to switch secondary coolant system low flow scram modes from the well water source mode to the city water source mode. This switchover is apparently a manual action; is it covered in the facility operating procedures? Is there any indication on the front of the reactor control console that informs the operator which secondary coolant source (and low flow reactor trip setpoint) is in effect? If not, please describe the administrative controls that make this information available.

7-6 Sections 7.2.3.4.1 and 5.2 describe a coolant flow switch in the return line of the primary coolant system to the primary coolant storage tank which will scram the reactor in case of loss of return flow. The switch serves as a backup to the primary coolant low flow reactor trip instrument in the fill line. What is the setpoint for the return line flow instrument? Is the surveillance frequency the same as for the flow instrument in the fill line?

7-7 Many older analog components have become obsolete and are no longer available; 'equivalent' replacement components are not always true replacements. How are replacement electronics components to repair the analog instruments and electronic circuit boards qualified? Following repairs, how are acceptance tests selected to certify that circuit boards and logic modules, and the equipment in which they are installed, are functionally operable?

8 ELECTRICAL POWER

8-1 Provide single-line drawing(s) depicting supply feed(s) and distribution of normal and emergency sources of AC and DC electrical power systems (for example: is the voltage supplied at the desired service levels (230V and 115V) or is a step-down transformer used? How is power distributed inside the facility? Is there a main distribution center (motor control center) or are there multiple/individual distribution panels with individual/separate feeds from outside sources?).

8-2 Section 8.2. The fail safe behavior of the reactor protection system and control blades was described. Upon loss of power, does the primary coolant system dump value also drain the primary system?

8-3 Describe the design features (e.g., design and location of electrical wiring) provided to ensure that electrical power circuits are sufficiently isolated to avoid electromagnetic interference with safety-related instrumentation and control systems.

8-4 Describe any needs for electrical power that may be required for placing/maintaining experimental equipment in a safe condition.

8-5 Section 8.3, Emergency Electrical System. This section states that no credit is taken for the back-up electrical diesel generator for safety analysis considerations. In the event of an extended loss of the normal AC power source, will operation of the emergency power source (Diesel Generator) be relied on to ensure the availability/operation of systems which provide for personnel safety, habitability of the reactor facility, reactor status instruments, and radiation monitoring systems?

9 AUXILIARY SYSTEMS

9-1 Section 9.2.1, New Fuel Storage. This section states that loading and unloading of fuel into (and out of) the reactor core will only be performed by 'qualified reactor operators and staff.' Define what 'staff' members are permitted to perform these functions (as defined in Appendix 14.1 TSs). If other positions are included as 'staff', what are the qualification requirements for these individuals.

9-2 Section 9.2.3, Bridge Crane. The bridge crane is described as a 3-ton crane. Are there any restrictions or safety factors for the crane which limits the actual load which can be safely handled? Briefly describe what preventive maintenance or inspections are performed on the crane to ensure continued safe operation. Are there any restrictions with regard to handling heavy loads over the core? What is the weight of the fuel transfer cask?

9-3 The criticality accident requirements of 10 CFR 70.24 are applicable to the UFTR. Please discuss how this regulation is met.

9-4 Section 9.2.2, Spent Fuel Storage. What is the temperature as a function of storage time for the dry-stored spent fuel in the storage pits?

9-5 Section 9.6.3, Equipment and Floor Drainage System. This section states that the reactor building floor drainage system is designed so that liquid effluents go directly to the hold-up tank. But the section also states that there are no drains leading directly to the hold-up tank. Please clarify.

10 EXPERIMENTAL FACILITIES AND UTILIZATION

10-1 Confirm that loss of AC power is considered during the experiment approval process.

10-2 Provide a current copy of UFTR SOP-A.5 (Experiments).

10-3 Section 10.2.6, Automatic Transfer System (Rabbit). Provide a more detailed description of the design and operation of the automatic pneumatic sample transfer (Rabbit) system and the administrative controls governing its use. Specific topics to be addressed include the size (diameter) of tube and rabbit, potential consequences of a stuck/immovable rabbit assembly and design features and/or administrative controls provided to preclude or mitigate this occurrence.

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11-1 Please provide calculations to show that doses to the reactor staff and members of the public from the production of normal gaseous effluents from reactor operations is acceptable. The calculations should be based on continuous reactor operation (unless you want to limit reactor operation by license condition) and should consider both argon-41 and nitrogen-16. Doses should be determined for staff members and the maximum exposed member of the public, at the closest residence to the reactor and at any other points of special interest (e.g., dormitories), if applicable.

11-2 Section 11.1.2.3.2, Ventilation. This section refers to a 200 to 1 stack dilution factor. Please discuss the basis of this factor.

11-3 Section 11.1.2.4, Health Physics Program. This section states that the Radiation Control Officer supervises the actions of the UFTR RSR Subcommittee. Please explain why the Radiation Control Officer, an ex-officio member of the subcommittee, has supervisory responsibility over the RSR Subcommittee and how that affects the independence of the subcommittee.

11-4 In this chapter there is no mention of the special nuclear material and byproduct material limits in your current license. Please confirm that you want to maintain similar limits in your renewed license.

12 CONDUCT OF OPERATIONS

12-1 Section 12.1, Organization. The organizational chart (Figure 12.2) contains many lines with arrowheads and a diamond-shaped "or" box that are not completely clear. Please show reporting lines by solid lines and communication lines by dotted lines. Also show reporting responsibilities by arrows.

12-2 Section 12.1.3, Staffing, and TS section 6.1.3, Staffing. 10 CFR 50.54(m)(1) requires that an SRO shall be present at the facility for three specified activities. For example, an SRO shall be present at the facility during recovery from an unplanned or unscheduled shutdown. The SAR and TSs both use the words "direction" rather than presence. Further, the SAR uses the wording, "documented verbal concurrence from a Senior Reactor Operator is sufficient." The use of "sufficient" rather than "required" when discussing the verbal concurrence of the SRO seems to imply that the SRO may give concurrence for recovery without being present at the facility. The intent is to have the SRO present and to document their concurrence with the restart. Please update the wording or explain why it meets the requirements of 10 CFR 50.54(m)(1).

12-3 Section 12.1.4, Selection and Training of Personnel. The selection of personnel should meet the guidance in ANSI/ANS 15.4-1988. This is quoted in the TS but the SAR cites ANSI/ANS 15.4-1977. Please correct.

12-4 Section 12.1.5, Radiation Safety. Does the radiation safety staff have the ability to raise safety issues with the review and audit committee or university upper management and do they have the clear responsibility and ability to interdict or terminate licensed activities that they believe are unsafe? If not, how does the radiation safety staff deal with activities they believe are unsafe?

12-5 Section 12.1.5.1, Reactor Safety Review Subcommittee, and TS 6.2 Review and Audit. A quorum is defined as at least three members. But the membership is defined as at least five members. If there are more than six members a quorum of three would be less than half. The quorum should be at least three and at least half, also with the operating staff not constituting a majority (to meet ANSI/ANS 15.1). Also the Radiation Control Officer is referred to as both a member and an ex-officio member. Please address.

12-6 Section 12.1.5.1, Reactor Safety Review Subcommittee, and TS 6.2 Review and Audit. The SAR and TS should specify that all reports and minutes of findings and recommendations of the subcommittee should be submitted to Level I management; and should also specify which Level I manager(s). Please address.

12-7 Please address how you meet the requirements of 10 CFR 50.54(i) or (l).

13 ACCIDENT ANALYSIS

13-1 Ad hoc criteria were used in the SAR to extrapolate the BORAX I and II results to the UFTR core. Is there any transient calculation that shows the excursion energy for the UFTR in a nuclear excursion? What is the predicted maximum fuel temperature in the most limiting nuclear excursion? Is there any requirement on the coolant void and temperature reactivity feedback such that the maximum excursion energy is limited to 32 MW-sec?

13-2 The staff believes that the design basis accident (or maximum hypothetical accident [MHA]) chosen for the reactor in the SAR is extremely unrealistic and conservative. The purpose of the MHA is to conservatively, but realistically, bound the worse case radionuclide release that could occur. The staff has accepted a core crushing accident as the MHA for an Argonaut reactor and NUREG/CR-2079 has analyzed this accident for a generic Argonaut reactor. However, the NUREG/CR-2079 analysis is highly conservative and could be made more realistic by considering items such as, decay following reactor shutdown and isotope plateout. Also, as explained in NUREG-1537, the accident dose limits found acceptable to the NRC staff for reactors initially licensed before January 1, 1994, has been 5 rem whole body and 30 rem thyroid for occupational exposure and 500 mrem whole body and 3 rem thyroid for members of the public. Please reevaluate your MHA or provide justification as to why the MHA presented in the SAR is realistic.

13-4 In Section 13.3.5 the urban boundary was set to a distance of 0.5 miles. Instead, doses should be determined for staff members and the maximum exposed member of the public, at the closest residence to the reactor and at any other points of special interest (e.g., dormitories), if applicable.

13-5 Appendix 13-1. The same ratio was shown in Equation 13A-1 and Equation 13B-4 and it was used to adjust the BORAX non-melting excursion energy for the UFTR. What is the significance of the ratio? Was it an indication of the heat capacity of the fuel plate (per discussion on p. 13-A.1) or an indication of the heat transfer capability of the fuel plate (per discussion on p. 13-B.5)?

13-6 Appendix 13-B. In the last paragraph on p. 13-B.2 it was stated, "... the reactor could operate in the absence of protective actions at an equilibrium power level about 10 times higher than its normal maximum with little or no net steam production." Does this statement apply to the current normal power level of 100 kW or the original licensed power of 10 kW?

13-7 Appendix 13-B. On p.13-B.2 the heat removal capacity of 107 kWth was based on an assumed outside air temperature of 0°C. A more realistic outside temperature would significantly reduce the heat removal capability of the reactor coolant system. Please state whether this is an appropriate assumption.

13-8 Appendix 13-B. In Section 13B.2, what is the reference for correspondence between the excess reactivity of 0.6% $\Delta k/k$ and the asymptotic period of 0.8 seconds?

13-9 Appendix 13-B. What is the source of Figure 13B-1? Has its applicability to UFTR been demonstrated?

13-10 Appendix 13-C. Are the constants a_1 and a_2 in Equations 13C-1 and 13D-1 defined in Table 13D-1?

13-11 Appendix 13-D. Decay Heat Effects. In Section 13D.2, what is the reference for the calculation of the unit thermal conductance between the fuel plate and the fuel box? What are the bases for the assumed 50% Al-Al contact. What are the bases for the assumed contact pressure and the thickness of the air wall?

13-12 Appendix 13-D. What are the bases for the assumed 50% air and 50% graphite in the wall separating the fuel box and the graphite?

13-13 Appendix 13-D. What is the temperature of the heat sink (graphite) and how is it justified?

14 TECHNICAL SPECIFICATIONS

14-1 Technical Specifications (TSs). Bases are given for many of the TSs as required by 10 CFR 50.36. Please ensure that the bases for the TSs can be traced back to an analysis in the SAR. It is not clear when some of your TSs are applicable. For example, is TS 3.2.1(4) required to be met at all times or is it a requirement to take the reactor critical? Please review all TSs and ensure that it is clear under what conditions the TS applied.

14-2 Definitions. Please review your definitions to verify that they are used in the TSs or documentation that supports the operation of the reactor. Consider if definitions that are not used in operation of the facility are needed.

14-3 Section 2.0, Safety Limits and LSSS. As noted in the guidelines contained in NUREG-1537, Part 1, Appendix 14.1, Section 2.1.3, "... For plate type fuel... the applicant should determine a fuel cladding temperature below which cladding damage (softening or blistering) can be precluded. The applicant should then establish a corresponding power level, reactor conditions, and uncertainties that limit cladding temperature below the damage limit."

In the introduction of Section 2.1 you have correctly described the purpose of SLs and identified the fuel cladding as the principal fission product barrier to be protected. The process variables chosen should be those that if exceeded will quickly threaten the integrity of the fuel clad. One of the reasons why if a safety limit is exceeded, the reactor must be shut down until approval for restart is given by the NRC is to ensure that the fuel clad was not damaged when the safety limit was exceeded. NRC has accepted an upper fuel temperature for aluminum-clad, aluminum matrix plate-type fuels of 530°C. Plate blistering, a possible forerunner of cladding failure, has been observed above this temperature. While fuel temperature would be the best process variable for the safety limit, the inability to measure this process variable leads to the need to use variables that can be measured and controlled. For reactors with forced convection flow, the staff has accepted controlling the process variables of reactor power, coolant temperature, coolant flow, and if credit was taken in the analysis, height of water above the core. Exceeding the limit on primary coolant resistivity does not lead to immediate fuel clad damage. The NRC staff accepts primary coolant resistivity being controlled as a limiting condition of operation. Please develop safety limits for reactor power, coolant temperature and coolant flow based on keeping fuel temperature limited to 530°C. Discuss the need for a safety limit on height of water above the fuel elements, and if justified, propose a safety limit. Justification of safety limits usually appears in Chapter 4 of the SAR and the accident analysis in Chapter 13 of the SAR usually forms the technical bases for the limiting safety system settings (LSSS) and the safety limits.

14-4 Safety Limits and LSSSs. For those LSSSs that protect safety limits, provide an analysis that shows that automatic protection at the LSSS limit will protect the safety limit considering process uncertainty, overall measurement uncertainty and the transient phenomena of the process instrumentation.

LSSS 2.2(6). This does not appear to be a process variable limit such as the flow rate. It appears to be an on-off condition that is better addressed as a LCO. Please justify this as an LSSS or move to the LCO section of the TSs.

LSSS 2.2(7). This does not appear to be a LSSS because it is not a limit on a process variable. This appears to be a LCO or design feature. Please justify this as a LSSS or move the requirement to a more appropriate section of the TSs.

LSSS 2.2(10). This does not appear to be a process variable limit such as the flow rate. It appears to be an on-off condition that is better addressed as a LCO. Please justify this as an LSSS or move to the LCO section of the TSs.

LSSS 2.2(11). This does not appear to be a LSSS because it is not a limit on a process variable. This appears to be a LCO equipment operability requirement. Please justify this as a LSSS or move the requirement to a more appropriate section of the TSs.

The full bases for the LSSSs should be presented in the appropriate sections of the SAR.

14-5 Section 3.1(2), Excess Reactivity. A statement is made in Section 13.1.1.1 of the SAR that the UFTR is not planned to contain more than about 1.2% Δk/k excess reactivity even when freshly loaded. Given that statement, please justify the need for a TS excess reactivity limit of 2.3% Δk/k.

14-6 Table 3-1 and 3-2. It is not clear if some of the safety system operability tests are testing the operability of the safety system feature of concern. How does loss of primary coolant flow show operability of the low inlet water flow? How does loss of primary coolant level show operability of the low water level in core safety system trip? How does the loss of shield tank water level show shield tank low water level?

14-7 Table 3-2. An operability test of the period and power channels is required following a shutdown in excess of 6 hours. What is the basis of the 6-hour time period. Does this apply if the reactor is secured? The table tests component or scram function. Do these tests confirm the scram function of the control rods or, as appropriate, the safety system trip function of the control rods and the dump valve?

14-8 TS 3.3(2), Reactor Coolant System. Please explain the purpose of the 6-hour reactor operation statement as related to primary coolant resistivity. Please explain why primary coolant pH is not controlled?

14-9 Table 3-4, Radiation Monitoring System Settings. The stack radiation monitor has a fixed alarm at 4000 cps. What hazard does a warning at this count rate represent? How are changes in the efficiency of the monitor with time or as components are replaced accounted for?

14-10 Section 3.6(4), Explosive Materials. The TS refers to "limited quantities" of explosive materials that may be irradiated. Please either propose and justify a quantity of explosives or discuss the basic restrictions that explosives must meet to be irradiated (e.g., irradiation container has ability to contain by a certain factor the energy released if the explosive is detonated).

14-11 Section 3.6(7), Fueled Experiments. The TS refers to "a limit should be established" on the inventory of fission products in fueled experiments. Please propose and justify an upper limit on the allowable fission product inventory.

14-12 Section 3.8 Fuel and Fuel Handling. LCO 3.8(3) and (4) prohibit reactor operation with failed fuel. Is the primary coolant surveillance described in 4.3(3) the only means of detecting fuel failure, or are there other indications used to provide a more rapid indication of failed fuel? If failed fuel were detected, how would the specific failed fuel assembly be identified?

APPENDIX C

UFTR OPERATOR REQUALIFICATION
AND RECERTIFICATION PROGRAM
JULY 2005 – JUNE 2007
RENEWAL PACKAGE

ATTACHMENT II

UNIVERSITY OF FLORIDA

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May 25, 2005

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Requalification Training Program

University of Florida Training Reactor, Facility License R-56, Docket No. 50-83

The current operator requalification and recertification program training cycle for the University of Florida Training Reactor as submitted with a letter dated June 6, 2003 is scheduled to end in June 2005. Therefore, we propose to renew the current plan with only minor changes to update to new dates reflecting the next two-year training cycle. The revised plan will be essentially the same as that currently being used. A copy of the revised "University of Florida Training Reactor Operator Requalification and Recertification Program Plan" dated May 25, 2005 is enclosed and will be effective from July 2005 through June 2007. It should also be noted that a significantly revised program will be implemented when the UFTR is relicensed per submissions in July 2002 as part of the UFTR relicensing submittal and revised technical specifications.

As usual, we plan to continue using this proposed program beyond the next two-year cycle; that is, we will automatically restart the same two-year requalification and recertification program training cycle beginning in July 2007 and again every two years thereafter. If you need further information on this plan or the proposed usage of it for all future two-year training cycles, please let us know.

Sincerely,

William G. Vernetson
Director of Nuclear Facilities

WGV/dms
Enclosure

cc: A. Adams, NRC
Reactor Safety Review Subcommittee

Sworn and subscribed this 25 day of May 2005

Notary Public

Terri L. Sparks
Commission # DD346498
Expires August 12, 2008
Bonded Troy Fain - Insurance, Inc. 800-385-7019

UNIVERSITY OF FLORIDA TRAINING REACTOR

OPERATOR REQUALIFICATION AND RECERTIFICATION TRAINING PROGRAM PLAN

JULY 2005 through JUNE 2007

Submitted by

Dr. William G. Vernetson
Director of Nuclear Facilities

Department of Nuclear and Radiological Engineering
University of Florida
Gainesville, Florida

MAY 25, 2005

OPERATOR REQUALIFICATION AND RECERTIFICATION

TRAINING PROGRAM PLAN

(July 2005 through June 2007)

1.0 GENERAL

A training program for the periodic requalification of UFTR operators shall be conducted in accordance with the requirements established by this document. The requalification and recertification training for UFTR personnel meets or exceeds the requalification training requirements established by 10 CFR 55 and the ANSI/ANS-15.4-1988 standard entitled, "Selection and Training of Personnel for Research Reactors."

The objectives of this program are to refresh in areas of infrequent operation, to review facility and procedure changes, to address subject matters not usually reinforced by direct use, and to improve in areas of performance or knowledge weakness. The Program is designed to evaluate an operator's knowledge and proficiency for his duties and to provide and assure retraining where necessary in order to assure improvement. Emphasis is on those subjects considered necessary for continued proficiency. In addition, the Program takes into consideration the specialized nature and mode of operation of the UFTR as well as the background, skill, degree of responsibility, and participation of certified personnel in related facility activities. The Program also reflects facility modifications and changes in procedures.

Responsibility for the administration of the program shall rest with the Director of Nuclear Facilities for the Department of Nuclear and Radiological Engineering and his/her duly designated representative. Requalification examinations shall be administered by one knowledgeable of facility operation and applicable subject matter.

All licensed and certified operators are required to participate in all phases of this program except where specifically exempted. Normally exemptions are allowed only for the individual responsible to produce and administer the examinations. Persons in training for an operator's license also participate in the requalification program. An operator receiving a license during a requalification period is required to complete only those portions occurring after the effective date of the license received.

The requalification training program effective at the UFTR shall consist of ten (10) component areas described in the following sections and listed in Table 1. The requirements that must be met in order to complete the requalification program successfully are delineated in these sections.

Table 1

Operator Requalification and Recertification Program Requirement Areas

1. Requalification Schedule
2. Lectures, Reviews and Examinations
3. Operations and Checkouts
4. Emergency Drills
5. Absence from Authorized Activities
6. Evaluation and Retraining of Operators
7. Certification
8. Requalification Documentation and Records
9. Requalification Document Review and Audit
10. References

1.1 REQUALIFICATION SCHEDULE

The UFTR requalification and recertification training program shall be conducted biennially and shall be followed by successive two-year programs. To assure that the program is effective, the various requirements should be executed according to the time schedules outlined in this program guide. The current two-year Requalification Training Schedule (July 2005 – June 2007) is contained in Appendix A of this Program Plan.

1.2 LECTURES, REVIEWS AND EXAMINATIONS

1.2.1 <u>Lectures</u>

The requalification and recertification training program is divided into the group of topics listed below in Table 2, for which preplanned training or preparation is scheduled. The schedule is set up so that the entire program covering the topics listed in Table 2 is completed over the two year period.

Table 2

Requalification Training Lecture Program Topics

1. Nuclear Theory and Principles of Operation

2. Design and Operating Characteristics

3. Instrumentation and Control Systems

4. Reactor Protection System

5. Normal, Abnormal and Emergency Operating Procedures (all procedures are covered once in the two-year period, independent of special training on significant changes and independent of emergency drills)

6. Radiation Control and Safety

7. Technical Specifications and Applicable Portions of Title 10, Code of Federal Regulations

8. Emergency Plan

9. Security Plan (including security response procedures)

Self-study methods are also considered to be an adequate and appropriate training method for the lecture program topics when learning objectives are properly measured by examination or documentation of expertise. Self-study methods are especially advised in combination with lectures.

1.2.2 <u>Examinations</u>

1.2.2.1 <u>Lecture Program Topics</u>

An examination shall be administered at the end of each lecture session listed in Table 2; each examination should be administered no later than four weeks after the lecture or review session. For designated cases, a final examination covering all topics in a series of lectures may be substituted for individual examinations. Results of the certified individual's evaluation from the examinations is used as one input to determine the operator's proficiency, weakness or deficiency.

Examination is encouraged but not required for training sessions given but not required by this program.

The individual responsible for developing the examinations for the requalification program may be exempted from the examination. This exemption should be rotated among the eligible staff members as appropriate.

1.2.2.2 Biennial Comprehensive Examination

A comprehensive requalification written examination shall be required for all operators on a biennial schedule. A lecture may be given prior to this examination but is not required.

1.2.2.3 Annual Operations Test

Each reactor operator and senior reactor operator is required to take an annual operations test to demonstrate operational proficiency and understanding of system responses. This examination is administered by a designated Senior Reactor Operator.

1.2.2.4 Annual Walk-through Examination

Each licensed Reactor Operator and Senior Reactor Operator shall demonstrate satisfactory understanding of the operation of the facility systems, operating procedures and license as well as facility procedure and license changes during an annual walk-through examination administered by a designated Senior Reactor Operator.

1.2.3 Fuel Handling

Practical training in fuel handling shall be conducted biennially. Prior to any refueling operation and/or fuel handling operation, a special training session shall be held discussing/practicing the required operations and reviewing procedures to assure proficiency of all personnel involved, including emergency actions. This training may be credited as the required biennial fuel handling practical training.

1.2.4 Procedure/Technical Specifications Changes

Any changes in procedures, technical specifications, regulations, as well as any change with safety significance to the facility shall be reviewed by every licensed operator. Any procedural changes will be distributed directly to all licensed reactor operators and discussed as needed. Furthermore, a written monthly report summarizing the activities in the reactor facility, including modifications, maintenance, results of calibrations and tests, as well as significant occurrences such as potential violations, failures of systems, etc. will be made available as required reading for all licensed operators.

1.2.5 Required Reading List

Documents, letters and memos pertinent to operational safety shall be maintained in the Required Reading List prior to permanent filing. Each operator is responsible for reviewing the list periodically and in a timely manner to remain current with the information contained in the Required Reading List. This reading list will be indexed with a master listing with spaces provided for initials of all required readers. This list should be reviewed at intervals not to exceed one month; when an item has been reviewed, the proper initials should be affixed to acknowledge completion of review.

1.2.6 Yearly Review

A yearly review of facility operations, maintenance, modifications, etc. is conducted with the operating staff by the Director of Nuclear Facilities or the Reactor Manager using the UFTR Annual Report as a basis for the review. More frequent reviews may be conducted as appropriate.

1.3 REQUALIFICATION OPERATIONS AND CHECKOUTS

1.3.1 Reactivity Control Manipulations

Over the two year requalification period, each certified individual shall perform at least ten reactivity control manipulations in any combination of reactor startups, shutdowns, or significant reactivity changes.

1.3.2 Schedule of Operations and Checkouts

To insure operator proficiency over a range of ordinary operations, the following schedule of operations and checkouts shall be maintained by all licensed operators when the reactor is operable.

1.3.2.1 Startups and Shutdowns

Each licensed operator shall perform at least one reactor startup quarterly at intervals not to exceed four months. This operation shall include at least one additional reactivity manipulation on a quarterly basis.

1.3.2.2 Daily Checkouts

Each licensed operator shall perform at least one daily checkout quarterly at intervals not to exceed four months.

1.3.2.3 Weekly Checkouts

Each licensed operator shall perform at least one weekly checkout semi-annually at intervals not to exceed eight months.

1.3.2.4 Quarterly Licensed Activities

To maintain certification, each licensed reactor operator shall exercise his/her operator's license for a minimum of four (4) hours of licensed activities during each calendar quarter.

1.3.2.5 Remediation Requirements

Any operator who fails to perform the required licensed activities listed in Section 1.3.2.1 through 1.3.2.4 must receive supervised practical training to meet each of these requirements prior to resuming solo operation for certified activities. In particular, if the requirement to exercise the operator's license for a minimum of four (4) hours of licensed activities during each calendar quarter is not met, then the license becomes inactive; prior to reactivation of the license (recertification), the Reactor Manager or alternate must verify that qualifications are current and the operator must perform six (6) hours of licensed activities under the direction of a licensed operator or senior reactor operator.

1.3.2.6 On-the-Job Training

The specific operational practices delineated in this Training Program Plan including the annual operations test, the annual walk-through examination, and the requirements for conducting facility checkouts, startups, shutdowns, reactivity manipulations including at least four (4) hours of certified activities per calendar quarter constitute the bulk of the operator on-the-job training requirements. In addition, the biennial fuel handling training as well as semi-annual training on emergency response equipment, quarterly emergency drills, and annual special equipment training are also considered a major portion of the practical on-the-job training and are considered adequate to assure safe operation of the facility.

1.3.3 Credit for Reactivity Control Manipulations

For the purpose of meeting minimum requalification and recertification requirements, other than the four (4) hours of licensed activities required per Section 1.3.2.4, each licensed operator may take credit only for reactivity control manipulations which they perform themselves. For senior reactor operators, direct supervision of these operations may be considered equivalent to actual performance.

1.3.4 Records

It is the responsibility of each operator to insure that Requalification Training Program's training requirements are met and logged in the operator's Requalification Notebook. Each operator shall also be responsible to ensure that monthly operating hours are logged in the same notebook.

1.4 EMERGENCY DRILLS

1.4.1 Scheduling and Participation

Emergency drills shall be held quarterly, per UFTR Technical Specifications Section 4.2.6(3). At least once per year these drills shall involve the participation of the University Police Department, the Gainesville Fire Department and other emergency assistance teams as appropriate for the drill in question. Each operator is required to participate in two emergency drills per year at intervals not to exceed eight months.

Any operator failing to meet this two-drill requirement must receive special training on proper response to emergencies and must receive a documented review of the last drill missed as well as a walk-through of the facility related to proper emergency responses. This remediation shall be conducted prior to performing certified activities.

1.4.2 Postdrill Critique

A review of the drill and applicable emergency procedures shall be performed with all certified individuals within 30 days after completion of the drill. This review should include any deficiencies as well as recommendations for improvement and is normally conducted immediately after the drill for all operators and other staff and radiation control personnel involved in the drill. Nonparticipating certified individuals may perform this review using the drill record in the required reading file or participate in a special training session. Documentation is provided via initials in the Required Reading List or on forms documenting special training sessions.

1.5 ABSENCE FROM AUTHORIZED ACTIVITIES

An operator who has not been actively performing certified functions for a period in excess of four months shall be required to demonstrate to the Reactor Manager or duly authorized representative that his/her knowledge and understanding of the operation and administration of the facility are satisfactory before returning to certified duties. This shall be accomplished through an interview and evaluation <u>or</u> a written, oral or operational examination <u>or</u> a suitable combination thereof. Any deficiencies uncovered must be corrected before the individual resumes performance of certified functions.

1.6 EVALUATION AND RETRAINING OF OPERATORS

1.6.1 <u>Grade Requirements</u>

The acceptance criterion on all graded examinations shall be 80%; all operators are required to complete each examination satisfactorily according to the following requirements:

1.6.1.1 A score on the written or other examinations equal to or greater than 80% may require no additional training. Nevertheless, the results of all examinations to include missed questions should be reviewed with the operator to assure proper understanding.

1.6.1.2 A score on the written or other examination in the range of 65%–79% requires additional training in those areas or topics where weaknesses or deficiencies are indicated. This retraining and retesting shall be completed within 60 days from the date the examination was administered and prior to the candidate being recertified. In this case the candidate need not be removed from licensed duties subject to the evaluation of the Reactor Manager or his/her duly authorized representative.

1.6.1.3 A score on the written or other examination of less than 65% requires that an evaluation be performed by the Facility Director or designated representative within one month. The evaluation shall determine if the deficiencies require that the individual's certification be withdrawn pending completion of any accelerated retraining effort. The evaluation shall take into account the individual's past performance record, the supervisor's evaluation, and past test scores as well as current deficiencies. Additional oral or operational examinations may also be given to aid in the evaluation. In any case certification shall be withdrawn within four months if the candidate does not achieve passing scores after reexamination.

1.6.2 Accelerated Training

Accelerated training programs shall be completed within four months following the grading of an examination. Furthermore, within one month after the grading of the examination, there shall be an evaluation by the Reactor Manager or a designated representative to determine if the deficiencies uncovered warrant withdrawal of the individual's certification pending completion of the accelerated training program. The evaluation shall consider the individual's past performance record, the supervisor's evaluation and past test scores as well as current deficiencies. Additional oral or operational exams may also be given to aid in the evaluation.

1.6.3 Additional Training Requirements

Additional training shall be provided whenever needed to correct weaknesses or deficiencies uncovered. Such additional training shall be completed prior to the conclusion of the specific requalification program or application for renewal of operator's license, whichever occurs first.

Additional appropriate training requirements in the form of formal lectures, tutoring, self-study or on-the-job training shall be based on the results of examinations conducted.

1.6.4 Deficiencies Affecting Safety

Regardless of the score, if the individual's test indicates a deficiency in a critical area that affects safety, training shall be promptly administered to correct the deficiency or the operator will be removed from performing certified duties in the affected area until the deficiency is corrected.

1.6.5 Evaluation Via Annual Examinations

The annual operations test and the annual walk-through examination are key factors in evaluating the continued competence of the certified operator both for demonstrating operational proficiency and understanding of system responses and for demonstrating overall satisfactory understanding of the operations of the facility, operating procedures and facility license changes. The results of these two examinations should be utilized as primary input for evaluating operator performance for recertification purposes.

1.6.6 Biennial Evaluations

An in-depth evaluation of the operating performance of each licensed operator shall be performed and documented biennially as a minimum by a summary and judgmental statements. The operational evaluation provides an estimate of the knowledge, competence and dexterity of the operator to operate the reactor safely and to take appropriate actions in response to abnormal and emergency situations that may arise. Additional operational training shall be provided to correct performance weaknesses that may be identified.

The biennial evaluation shall include results from the written examinations, the annual operations test, the annual walk-through examination and other on-the-job evaluation of operational proficiency as well as any other available indications of the operator's capability to discharge his/her duties in a safe and competent manner including participation in practical and special training, instructional activities and other work activities.

1.6.7 Additional Evaluations

An evaluation shall be made of an operator at any time his/her physical or mental condition appears impaired in a manner that his/her performance of duties as an operator appears to be affected. Any exemplary performances or additional duties performed by an operator should be noted in his/her Requalificational Folder/ Notebook to aid later evaluations.

1.7 RECERTIFICATION

1.7.1 Certified individuals who have successfully completed the requalification program may be recertified by the Facility Director or designated alternate.

1.7.2 All certified individuals must be cognizant of facility technical specifications, design and procedure changes in a timely manner.

1.8 REQUALIFICATION DOCUMENTATION AND RECORDS

1.8.1 Operator Requalification Records

Operator requalification records shall be kept to assure that all the requirements of the "UFTR Operator Requalification and Recertification Program Plan" are met.

Each operator shall have an individual folder or notebook containing signature blocks for lectures attended, prepared or assigned self-study sessions, reactivity manipulations performed, weekly and daily checkouts performed, and quarterly drills participated in by the operator. The notebook shall also contain copies of written examinations administered, the answers given by the operator, results of any evaluations and documentation of any additional training administered in areas in which an operator has exhibited deficiencies. The performance of, or participation in, special training such as for fuel handling, use of emergency equipment, crane operation, etc., should also be logged in the applicable Requalification Notebook.

1.8.2 <u>Requalification Training Manual</u>

A Master Requalification Training Manual will be used to organize training requirements; this manual shall contain a schedule of all required lectures, reviews, emergency drills, and other exercises. The date the item is performed shall be indicated in this manual. A section of this manual shall be designated to contain completed training items, attendance sheets, master copies of tests given and lecture outlines if available.

A separate section of this manual shall also indicate operator license amendment commitments and the dates for each including relicense dates for all licensed operators.

1.8.3 <u>Records Retention</u>

Required documents and records pertaining to the Requalification and Recertification Program shall be maintained at the UFTR as part of the facility records for at least six years. Per 10 CFR 55.59(5)(i), these records including the master training file shall be retained for each reactor operator or senior reactor operator until the respective operator's license is renewed or surrendered.

1.9 REQUALIFICATION DOCUMENT REVIEW AND AUDIT

The individual Requalification Folders or Notebooks shall be reviewed on a semi-annual basis, at intervals not to exceed eight (8) months, by a designated Senior Reactor Operator and shall be noted by the inclusion of the SRO's dated signature. Any deficiencies noted during the review shall be brought to the attention of the Director of Nuclear Facilities or the Reactor Manager who will then insure that appropriate corrective action is taken.

An audit of requalification program records shall be conducted by the Reactor Safety Review Subcommittee (RSRS) biennially at intervals not to exceed thirty (30) months. Such an audit should be performed annually at intervals not to exceed fifteen (15) months. All such audits shall be documented by the RSRS via its audit report or equivalent document.

1.10 REFERENCES

1.10.1 Title 10 Code of Federal Regulations, Part 55, "Operators' Licenses."

1.10.2 American National Standard ANSI/ANS-15.4-1988, "Selection and Training of Personnel for Research Reactors."

APPENDIX A

UFTR REQUALIFICATION

TRAINING PROGRAM SCHEDULE

2005-6 UFTR REQUALIFICATION TRAINING SCHEDULE

JULY	AUGUST	SEPTEMBER	OCTOBER	NOVEMBER	DECEMBER
	(L) Design and Operating Characteristics	(P) EMERGENCY DRILL	(P) Emergency Equipment Training	(L) Nuclear Theory and Principles of Operation	(P) EMERGENCY DRILL (involves outside agencies as appropriate)
			(P) Special Equipment Training (Rabbit System, Overhead Crane)		(L) Security Plan
					(I/P) Annual Operations Test
JANUARY	**FEBRUARY**	**MARCH**	**APRIL**	**MAY**	**JUNE**
	(L) Normal, Abnormal and Emergency Procedures	(P) EMERGENCY DRILL	(L) Reactor Protection System	(I) Operator Walk-through Exams	(P) EMERGENCY DRILL
	(P) Fuel Handling Training		(P) Emergency Equipment Training		
	(S) Annual Report Review				

(P) = PRACTICAL TRAINING (S) = STAFF TRAINING (I) = INDIVIDUAL TRAINING (L) LECTURE/EXAM

2006–7 UFTR REQUALIFICATION TRAINING SCHEDULE

JULY	AUGUST	SEPTEMBER	OCTOBER	NOVEMBER	DECEMBER
(L) Instrumentation and Control Systems	(L) Radiation Control and Safety	(P) EMERGENCY DRILL	(L) Technical Specifications		(L) Emergency Plan
			(P) Emergency Equipment Training		(P) EMERGENCY DRILL (involves outside agencies as appropriate)
			(P) Special Equipment Training (Rabbit System, Overhead Crane)		(I/P) Annual Operations Test

JANUARY	FEBRUARY	MARCH	APRIL	MAY	JUNE
	(I) Operator Walk-through Exams	(P) EMERGENCY DRILL	(P) Emergency Equipment Training		(P) EMERGENCY DRILL
	(S) Annual Report Review				BIENNIAL COMPREHENSIVE EXAM

(P) = PRACTICAL TRAINING (S) = STAFF TRAINING (I) = INDIVIDUAL TRAINING (L) LECTURE/EXAM

<u>APPENDIX D</u>

CORRECTION TO 2003–4 UFTR ANNUAL PROGRESS REPORT

Chapter II
University of Florida Personnel
Associated with the Reactor

(page II-1 is resubmitted to correct a date reference
in footnote 1 to indicate August 8, *2003*)