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THESIS

WRIGHT-PATTERSON AIR FORCE BASE, OHIO

AF-WP-B-SEP 60 5M

A NUCLEAR REACTOR FACILITY SAFETY INSPECTION GUIDE

THESIS

Presented to the Faculty of the School of Engineering of
Institute of Technology
Air University

in Partial Fulfillment of the
Requirements for the
Degree of Master of Science

By

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GNE 61

February 1961

Preface

I am deeply indebted to the many individuals and organisations who have contributed to this study. Major James R. Bohannon, Jr. Associate Professor of Physics of the Institute of Technology, recommended the topic, and provided initial information and continuous encouragement and guidance. This assistance from Major Bohannon, my adviser, was of immeasureable value.

The study was greatly enhanced by Dr. Peter Morris, Compliance Division, Headquarters, Atomic Energy Commission, who generously provided his draft copy of an A.E.C. inspection guide. Lt. Colonel Rudolph W. Staffa and Major William C. Burns, of the Nuclear Power Division, U. S. Army, contributed information concerning current Army inspection procedures which was extremely helpful. In addition, Mr. Francis X. Gavigan of the A.E.C. Chicago Operations Office provided a very fine insight into the procedural details of the A.E.C. inspection system. Mr. Valta G. Lewis, Operations Officer of the NETF at Wright-Patterson AFB, offered a very constructive review of the detailed inspection guide for operations and maintenance. Finally, Lt. Colonel Claude DeLorenzo of the DNSR, Kirtland AFB, New Mexico reviewed the draft form of the thesis and contributed many helpful comments. I am deeply indebted to all of these individuals for their courtesy and kind assistance.

It has been my intent to prepare a study which will serve as a guide to people responsible for conducting a safety inspection of a nuclear reactor facility in general, and Air Force reactor facilities in particular. The attention to detail is intended to explain and be indicative of the significance of nuclear facility safety inspections, and to provide a degree of standardization of the inspection itself.

Finally, my gratitude is hereby formally extended to my wife, Linda, for her constant encouragement and endless patience.

Leon E. McKinney

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Abstract

The purpose of this thesis is to propose a detailed safety inspection guide for use by the Air Force in inspecting its nuclear reactor facilities. Even though nuclear reactors ard systems should be designed so as to be inherently safe, it is not possible to attain a 100 per cent foolproof design. Thus it is necessary, in view of the magnitude of potential loss to be suffered from a nuclear reactor accident, to inspect reactor facilities for safety. The Air Force is responsible for safety at USAF reactor facilities and has initiated, through the Directorate of Nuclear Safety Research (DNSR), a periodic inspection program. It is toward this program that this guide is slanted.

In order to logically lead into the detailed inspection guide, the philosophy of inspection is discussed, both in general terms and in specifics dealing with reactor facilities. The conduct of an inspection is discussed, along with the importance of the human element in nuclear reactor safety. In order to better effect an understanding of the problem of safety inspection, inspection procedures currently in use by the A.E.C. are explained as well as self-inspection programs employed by contractors.

The present Air Force organizational structure for its safety inspection program and the various inspections of a reactor facility through its first year of operation are discussed in detail. Then, proposed procedures with which the Air Force could effect its facility inspection program are put forth and discussed at some length.

The heart of this report, the detailed inspection guide for the areas of operations and maintenance at a reactor facility, deals with the problem in a chronological sequence. Pre-inspection actions to be taken by an inspector are discussed, emphasizing that the facility hazards summary

should be closely studied prior to the inspection. The inspector confers with the operations supervisor to obtain information on the procedures, policies, and organization of the facility.

The inspector then proceeds to the reactor operations area and determines from contact with the operating personnel the duties and performance standards maintained to enhance the safe operation of the reactor. This entails interviewing the operations manager and operating crews. A checklist provides questions which are intended to establish the qualification level of the operating personnel. The next phase concerns the observation of the operators and the reactor during startup, power and period scrams, interim startups, and finally a normal shutdown. During this phase and after shutdown, the reactor system is inspected, from the operations standpoint, for safety aspects of the controls and instrumentation, heat transfer problems, safety systems, shielding and radiation control, operational procedures and records, and experimental systems. Finally, the checklist specifies questions for inspection of the facility maintenance program.

Two appendices are provided to properly complete the report. An abbreviated inspection guide is provided which covers the safety inspection of the entire facility. Finally, in order to emphasize the importance of a sound inspection program and to illustrate examples of how accidents occur, six nuclear reactor accidents are discussed and analyzed.

A

NUCLEAR REACTOR FACILITY

SAFETY INSPECTION GUIDE

I. Introduction

Purpose.

The purpose of this thesis is to propose a detailed safety inspection or survey guide for use by the Air Force in inspecting its nuclear reactor facilities. This guide, along with supplementing material, should help standardize the methods and philosophies of conducting inspections of a nuclear reactor facility. The term, facility, includes the reactor and its ancillary equipment, the site on which the reactor is located, and the organization which operates the facility.

Necessity for Inspection.

Since nuclear reactors are designed and constructed in such a manner as to be almost inherently safe, it might be asked, why is it necessary to inspect these facilities? Doctor Edward Teller, renowned physicist, clearly stated one reason. "With all the inherent safeguards that can be put into a reactor, there is still no foolproof system. Any system can be defeated by a great enough fool. The real danger occurs when a false sense of security causes a relaxation of caution." (Ref 1: 135-157) Assuming, then, that man is incapable of idiot-proofing a nuclear reactor, and that inspections are necessary to achieve safe conditions, how important is nuclear reactor safety?

The case for nuclear reactor safety is so strong that it cannot be overstated. Its importance is emphasized by the rigid standards and controls placed on the United States nuclear program by Title 10, Code of Federal Regulations. The attitude of the Atomic Energy Commission and American industry on nuclear safety is typified by the immense amount of material published on the subject. For example, Reactor Safety, A Selective Bibliography, published by the A.E.C. in 1959, lists over four hundred references in this field. The consequences of a serious reactor accident would be so dire in terms of health damage, financial loss, and loss of public faith that all possible action must be taken to minimize the possibility of such an accident.

Air Force Responsibility.

In keeping with this fact the Air Force is responsible for insuring that proper safety conditions exist and are maintained at all of its nuclear reactor facilities. The Air Force responsibility area is outlined in section 91b of the Atomic Energy Act of 1954, and additional details are given in chapter III under the present organization of the Air Force for nuclear safety. The basic task is, however, to establish optimum safety conditions at Air Force facilities. This means that the potential hazards to both the public and the operating personnel will be minimized. The potential hazards might be listed as follows:

1. Supercritical Excursion or runaway.
2. Release of radioactive effluent from the stack.
3. Ground seepage of radioactive material from storage tank.
4. Radioactive waste disposal.
5. Excessive radioactivity near the reactor due to abnormal operating conditions.
6. Fuel storage.
7. Potential release of radioactivity or energy through an act of God, aircraft crash, or a similar cause.
8. The normal radiation environment in day-to-day operations (Ref 47).

In order to insure that these potential hazards do not materialize, Air Force periodically inspects its nuclear facilities. The team conducting these inspections should consist of personnel who are absolutely qualified, by reason of their background and experience, as competent safety inspectors (see chapters II and III for additional information on the inspection team). This team represents the Directorate of Nuclear Safety Research (DNSR) which answers directly to the USAF Deputy Inspector General for Safety. The DNSR is responsible for safety at all Air Force reactor facilities, and is comparable to the Reactor Inspection Branch of the A.E.C.. The Air Force plans to conduct five types of inspections (see chapter III), but only two of these are discussed in detail in this guide. These are the operational reviews, or periodic inspections, and the special review or study.

<u>Outline of Report.</u>

The nuclear reactor facility discussed herein is assumed to have been in operation for a period of one year or more. Further, the type of reactor is not specified in order that the inspection guide might be more generally applicable, since the Air Force is operating a variety of reactor types. The discussion of inspection techniques and supporting data is, however, explicit. Time limitations did not allow a detailed discussion of all functional areas of a reactor-complex inspection. Thus the areas of operations and maintenance were selected for detailed discussion and support. These areas were chosen since it was felt that they were of primary importance and might be most influenced and improved by inspection. They are indeed the primary contributory source of hazards, and the principles of inspection applied here would be appropriate to inspection of other complexes.

In order to present a true perspective of the problem of inspecting a reactor facility, this report includes the following sections:

1. Philosophy of inspection.

2. Current and Proposed Inspection Procedures.

3. A Detailed Safety Inspection Guide-Operations and Maintenance.

4. Summary and Recommendations.

An abbreviated reactor facility inspection guide which covers the entire facility is included as an appendix for ease of reference. In order to emphasize the reality and necessity of reactor inspection and to illustrate how accidents occur, a second appendix discusses and analyzes six nuclear facility accidents. Finally, following the regular bibliography, a supplementary bibliography is included, in order that the reader might be guided to the best of the many additional sources of detailed information in the field of nuclear reactor safety and inspection.

II. Philosophy of Inspection

General Introductory Comments

The inspection of equipment and personnel has been vital facet of military operations from the earliest times. Commanders at all echelons have relied on the inspection to inform them of the status of men and machinery and to motivate men to maintain themselves and their equipment at prescribed standards. This basic philosophy is applicable to inspection of nuclear reactor sites today, with one modification. The safety inspection of nuclear reactor facilities is primarily a preventive type inspection, rather than an operational efficiency inspection. The importance of safety inspections is magnified by the astronomical magnitude of the potential hazards. The inspection methods can be broken down into two major areas, equipment and operating personnel, but it must be kept in mind that the two are really inseparable. Lest one should minimize the importance of the personnel who operate and administer a reactor site, due to the sophistication of safety and control equipment, it might be wise to consider a comment by Marvin Mann, a long-respected authority in the field of reactor inspection.

> "In its simplest terms the inspector's job is to check out the machine and the people who run it. The Facility and its design determine the potential for hazard. Depending on the people, the potential hazard can be (1) minimized, (2) unchanged, or (3) realized". (Ref 2: 27)

How to conduct an inspection

The methods used by the inspector will and should vary with the circumstances. Nevertheless, it is necessary that some uniformity exist, in fairness to the licensees ("operating group") and to permit the evolution of an orderly and reasonable program for reactor inspection within the Directorate of Nuclear Safety Research. For this reason a checklist is provided in Appendix A as an aid to reactor inspection. It is suggested that this checklist be used before an inspection as an aid in planning the actual work of the the inspection, and as an aid in writing the report after the inspection. (Ref 3: 8) Further benefit might be gained from distributing such a checklist to all Air Force reactor complex management chiefs. This would afford these personnel with a safety guide which they might use for year-round operations.

It must be kept in mind that the primary function of the inspection team is to promote safety in the operation of Air Force nuclear facilities. Safety is very definitely a function and responsibility of management, and although not their sole responsibility, it is one in which management must be seriously interested. The manner in which the inspection team conducts itself directly affects the responsiveness of management and operating personnel to the comments made by the inspectors.

"For example, at the beginning of the first visit to a facility, it is desirable to arrange a meeting with top management to explain the function, responsibility, authority, organization, and policy of the inspection team." (Ref 3: II-1) This forthrightness should help to clear the air of suspicion and skepticism and enable the inspection team to obtain the cooperation of the personnel being inspected. Further, the professionalism of the team will go far in making the inspection worth while to the operating group being inspected. The inspectors should then proceed in such a manner as to clearly evince the fact that their primary job is, as stated above, to promote safety at the facility. Inspections are to aid, not to hinder!

"As representatives of the Air Force Directorate of Nuclear Safety

Research, inspectors must maintain objectivity and impartiality. This implies that criticism or recommendations regarding kinds of equipment or systems are not in order unless directly related to safety. On the other hand, to promote safety, the inspector has an obligation to spread information and to review experiences wherever he goes. This must be done with unambiguous identification of personal opinion and predilections". (Ref 3: II-2) Such comments should be made in a tactful manner, with no trace of condescension. In all cases, when practical, at the conclusion of a visit, a review or critique of the inspection should be given to those personnel designated by the management. (Ref 3: II-1) Such a practice will greatly enhance the value of the inspection, as the findings will be much more apparent while yet fresh in the memory of the inspected personnel. In addition misunderstandings should be corrected on the spot.

It is highly desirable, if not mandatory, that the inspection be accomplished by men who have extensive experience in the operation of reactors. Furthermore the inspector should have some experience in experimental and theoretical investigations as well as preparation of design criteria, if possible. In general the reactor inspector will not need to exercise judgment concerning established standards (such as ASME codes), but only will ascertain that compliance with such standards is maintained. If circumstances should warrant, specialists or consultants may be used to study the particular features of a reactor program. (Ref 3: 7-8)

Effect of Human Element

As has been previously pointed out, it is not possible to design a reactor in such a manner that all potential hazards are eliminated. The competence and attitude of the personnel operating the reactor complex determine to a great extent the degree of safety attained. It is for this reason that emphasis is placed on inspecting operating personnel, safety organizations, the attitude of management, and organizational procedures, as well as the reactor and its ancillary equipment.

The author, even with his limited experience, has observed several occasions in which an obvious safety violation occurred around an operating or test reactor. In every case, the person or persons acting contrary to sound safety practices did so through an apparent feeling of overconfidence or even of contempt toward the potential hazard involved. All of these people were highly educated and trained, and all had a great deal of experience with reactors. It is much better for an inspector to point out these flaws than for the offender to be made aware of such malpractices through an accident.

Perhaps the final answer to this problem of overconfidence and carelessness is suggested by M.A. Schultz in commenting on a paper presented to the American Nuclear Society by Marvin Mann.

"I would like to emphasize the point that Dr. Mann made on written procedures. It seems to me that industry has now grown up to the point where we are not going to operate reactors with physicists, but rather large reactor installations will be run by disciplined high school graduates. This condition will occur for two reasons:

1. Because I really believe that high school technicians make better operators than physicists.
2. It is somewhat difficult to get a graduate physicist to work third shift.

What we really are seeking is an operator with enough disciplined training to follow procedures explicitly, yet with enough intelligence to recognize an abnormal situation when he sees it. Even in an abnormal situation, procedures must be followed - even if the rules only call for shouting for the boss. Large reactor operation will be routine most of the time. Anyone who gets bored with this situation is apt to be dangerous. Written procedures rigidly followed are our best safety protection". (Ref 4) This point is emphasized all the more when one considers that use of written checklists is mandatory for Air Force pilots during startup, cruise, and shutdown. A nuclear reactor is certainly more lethal than an aircraft!

III. Current and Proposed Inspection Procedures

Introductory Comments

The need for safety inspections of nuclear reactor facilities having been emphasized, and the basic philosophies of inspections having been outlined, the question arises as to how have reactor facilities been inspected in the past fifteen years. Assuming that the Atomic Energy Commission and industry have profited from experience gained over this period, one might better investigate how inspections are presently being conducted. This analysis is broken into three major sections: the first deals with the inspection methods used by the Atomic Energy Commission, the next with those methods used by the contractor or industry, and finally, the existing organization in the Air Force and proposed inspection methods.

In order to advance a more comprehensive understanding of the inspection guide and the detailed checklist, the complete inspection cycle, from reactor design thru operation, is discussed in the A.E.C. inspections methods section. This will enable the reader to understand the various stages of inspection that will have preceded the inspection of the reactor discussed in this guide, as said inspection occurs after the reactor has been in operation for a period of one year. The discussion of the safety program employed by industry, or contractors, will further improve an understanding of the total safety program.

The Inspection System of the Atomic Energy Commission

Organization of AEC for Inspection. The Atomic Energy Act of 1954 assigned to the Atomic Energy Commission the general responsibility for inspection of nuclear facilities in the United States. As will be pointed

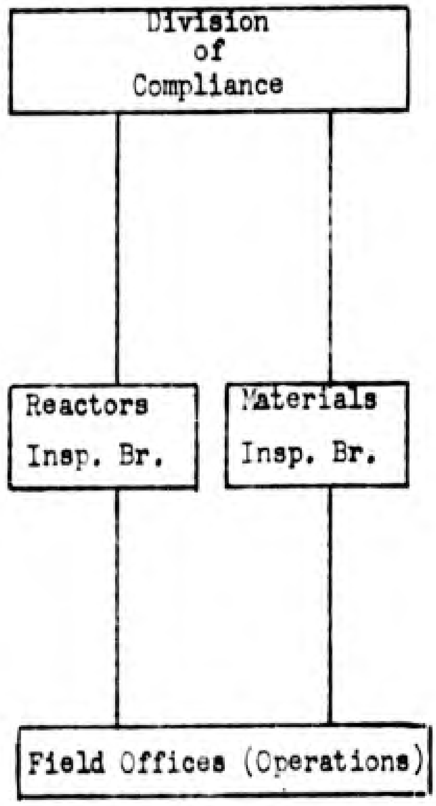

Fig. 1

AEC Organization for Compliance and Inspection (Ref 5) out later, there are certain exceptions to this blanket statement. The AEC fulfills its inspection responsibilities through an organization which is detailed in Figure 1. Actually the AEC conducts two types of inspections. (Ref 5) First are the inspections of facilities which are not operated by the AEC, but are operated by a contractor or by civilian industry. Second are those inspections which the AEC conducts at facilities operated under AEC control by civilian industry through AEC contracts. Although the general inspection techniques are almost identical, there are certain important differences which bear discussing.

Inspection of non-AEC Facilities. "Reactor inspections will differ widely, depending on the type of reactor, the state of construction or operation, the experience and ability of the licensee, and so on. For a reasonably large reactor or a reactor of new design, the inspection function will involve the following phases:

1. Familiarization of inspection personnel with the facility during design review and hazards evaluation activities.
2. Observation and review of construction progress.
3. Observation and review of tests of equipment and procedures before nuclear operation.
4. Observation and review of the initial loading of fuel, initial reactor startup and operational program". (Ref 3: 4)

The goal in the first phase is to become familiar with the facility design and safety features. During the construction phase, features and equipment peculiar to nuclear reactor facilities which pertain to safety are inspected in detail. Conventional types of inspection (compliance with welding, electrical, and building codes) are not performed by the A.E.C., but a check is made to see that such inspections are properly made. In the third phase, the inspector decides on whether the construction permit and license application have been adhered to, and whether the licensee will operate the facility in a sufficiently safe manner. On the basis of this inspection the A.E.C. is able to decide, prior to issuing a license, whether the facility has been completed in accord with the construction permit and license application. During the fourth phase, operation of both equipment and personnel is observed and evaluated. Finally, the fifth phase consists of nearly the same inspections as in the previous two phases, scheduled as frequently as is deemed advisable by the Division of Inspection. (Ref 3: 6-7)

"At the present time, responsibility for reactor inspections is held by the Division of Inspection Headquarters. However, reactor inspections will continue to be done by field personnel as well as headquarter personnel, according to the policy and schedules developed by the headquarters staff. The inspector is responsible for planning, carrying out and reporting the results of any of his reactor inspection activities described herein. He is responsible for examining the state of compliance, or lack thereof, of the licensee, collecting evidence to support any findings made and evaluating the safety of the operation. He is authorized to communicate directly with the Division of Inspection headquarters, concerning items requiring immediate action. Officially, he has no authority to suggest or recommend action on the part of a licensee". (Ref 3: 9-10) During an inspection of a non-AEC facility, of course, the inspectors do not discuss their observations with the people being inspected. (Ref 5)

<u>Inspection of AEC Facilities</u>. Basically the methods presently in use by the AEC for inspecting its own facilities are only slightly different from those outlined in the previous section on inspection of non-AEC facilities. These differences are worth noting, however, for the self-inspection

situation is somewhat analogous to that existing in Air Force facilities. The Air Force inspection team, representing the Directorate of Nuclear Safety Research, is comprised of Air Force personnel, and inspects facilities operated explicitly for the Air Force and, perhaps, by Air Force personnel. Similarly, the A.E.C. facilities are operated for the Atomic Energy Commission, and in some cases, by A.E.C. personnel. It should be pointed out that the inspection responsibility in the Air Force is given to the DNSR, an organization independent of operating responsibility.(Ref 47) This, of course, does not mean that the inspections are not seriously performed, or that the results of the inspection are not carefully heeded. In so far as inspection techniques are concerned, the major deviation in the AEC self-inspection system is that the inspectors might make on-the-spot comments so that immediate corrective action may be instituted, where applicable.

Methods used by Contractors

General. The inspection program generally employed by the contractor, or the company which is operating the reactor facility, is not an inspection program in the sense that the AEC and the Air Force inspection programs are. This is rightfully so, for an actual inspection of an operating facility is necessary only on a periodical basis, perhaps yearly, as outlined in the AEC program. The contractor actually employs a total safety program which operates continuously throughout the year. Such a program should include radiation safety, ground safety, industrial safety, and similar phases.

The contractor implements his program through a system of safeguards. These consist of an administrative structure, careful supervision of all activities, a program of continuous and periodical monitoring of specific areas and personnel, and a training and indoctrination program. There are several nuclear laboratories and reactor facilities operated for the Air Force by civilian contractors that exemplify contractors that closely adhere to such a program. (Ref 6: 49)

Organization and Administration. It is desired that company policy establish safe operation of the nuclear facility as a responsibility of line supervision. (Ref 6: 49) This is a very important requirement, for the company must realize and emphasize the importance of safe operations or no safety program can succeed. In other words, a successful safety program depends not only on the safety committee, but on line supervision, which starts with the chief executive and goes down through the supervisor who has one man working for him. Most contractors do establish a safety committee comprised of operations, engineering, reactor physics, health physics, and nuclear safety personnel.

Generally, such a safety committee will be responsible for reviewing all phases of operations of a nuclear facility from the nuclear safety standpoint (see chapter IV, Safety Committee section, for detailed responsibilities). It will accomplish this function by monitoring the facility operation, to include all experiments and tests, to insure adherence to previously established procedures. In addition, the safety group will investigate new activities to determine their safety feasibility and formulate standard procedures, where necessary, to cover such activities. (Ref 7: 17)

Some facilities further their safety efforts by conducting intensive technical training programs for reactor operators. Regular academic sessions are held, with examinations, in order that the level of technical proficiency of the operators might be raised. To achieve active participation of all reactor operators, some contractors have made the academic training an integral part of personnel promotion policies. In addition, an informal cross-training program is used to further advance technical proficiency and working knowledge of the reactor. (Ref 6: 50-51) Such programs, properly backed by top line management, contribute to a contractor's self-inspection and self-analysis program.

Proposed Inspection System for USAF

Present Organization. USAF has safety responsibilities for nuclear facilities developed and/or operated by the Air Force under the terms of section 91b of the 1954 Atomic Energy Act, and for facilities developed for USAF by the A.E.C. when such facilities are to be tested or operated on USAF installations. (Ref 8: 1-2) "The Deputy Inspector General for Safety, Headquarters, USAF will be responsible for over-all Air Force policy guidance in the development of and compliance with Nuclear Reactor System Safety Studies, Rules, Reviews, and Surveys". (Ref 8: 7)

Implementation of this responsibility is accomplished through two primary groups, listed below:

1. Directorate of Nuclear Safety Research (DNSR).
2. Nuclear Reactor System Safety Group (NRSSG).

"Acting as a field extension of the Deputy Inspector General for Safety, the DNSR will:

1. Supervise the activities of, chair and provide the secretariat for the USAF NRSSG.
2. Conduct operational reviews for each reactor system or facility.
3. Determine the need for, and conduct the Inspector General's special reviews or special studies; distribute the resulting reports to the appropriate Air Force commands and agencies and to other governmental agencies.
4. When appropriate, request the participation of the AEC and other

governmental agencies and the action and using agencies in an operational or special review.

5. Determine the need for, and conduct Nuclear Safety Surveys; distribute the survey reports to the appropriate Air Force commands and agencies.

6. Distribute Safety Study Reports, after Headquarters, USAF approval, to action and using agencies, the Atomic Energy Commission and other appropriate agencies". (Ref 8: 8)

"Acting as an advisory group to the Deputy Inspector General for Safety, the NRSSG will assure the adequacy of safety features in nuclear reactor systems to prevent accidental or unauthorized reactor excursions and radiation exposures which will affect the safety and health of operating personnel and the general public. To accomplish this, the NRSSG will:

1. Review the Site Evaluation Safety Study, the Initial Safety Study, the Pre-Operational Safety Study, and other studies as necessary, of each nuclear reactor facility or system for which the Air Force has an operational or developmental responsibility.

2. Prepare and submit the resulting NRSSG reports, based on the above studies, to DNSR for review, comment, and transmittal to DIG/Safety for Headquarters, USAF action. (Recommendations in these NRSSG reports are directive after they have been approved by Headquarters, USAF)". (Ref 8: 8-9)

It might be pointed out that the Deputy Inspector General for Safety is analogous to the head of the A.E.C. Division for compliance. The NRSSG fulfills the functions of the Hazards Evaluation Board and the ACRS, whereas the DNSR is somewhat parallel to the Reactor Inspection Branch.

Proposed Inspection Procedures for USAF

Types of Inspection. The Air Force plans to conduct five types of inspections or studies to insure nuclear safety for all USAF nuclear reactor facilities:

1. Site Evaluation Safety Study.
2. Initial Safety Study.
3. Pre-operational Safety Study.
4. Operational Reviews.
5. Special Reviews or Studies. (Ref 8: 3-6)

For purposes of this guide, only the latter two types will be discussed as the reactor facility in question is assumed to have been in operation for a period of at least one year.

The operational review will be conducted to re-examine the adequacy and suitability of safety features in nuclear reactor system design and the procedures followed in operational use, and to examine the adequacy of the safety rules. The review, or inspection, will consist of an examination of the operational history of the facility and a field inspection of the facility. (Ref 8: 5) It is this type of inspection with which this report is primarily concerned. The basic charter for the inspection is the Hazards Summary, which can be used as a guide for establishing the basic compliance requirements, along with operating manuals, standard operating procedures, instrumentation manuals, and previous safety studies or survey reports.

In addition to operational reviews, the Air Force will also conduct special reviews or studies when necessary. Such a special review or study might be prompted by the discovery of unsafe conditions on a reactor of the type used at an Air Force facility, or by a request for modifications that affect nuclear safety. In addition, a special review could be initiated to determine if the responsible persons are aware of new developments pertinent to their reactor types. (Ref 8: 6)

Inspection Team. The success of the inspection depends largely on the qualifications of the individual members of the team and of the team as a group. The optimum condition would be that every team member have extensive experience in both the design and operation of reactors, to include breadth, depth, and length of experience. (Ref 3: 7) It is not necessary that the inspector have experience in conventional aspects of design and operation, such as ASME codes, for he only ascertains compliance with such conventional standards. Inspectors should be encouraged to call in consultants, without fear of embarassment, whenever any doubt arises as to the compliance with conventional standards. (Ref 3: 7-8)

There are several possible organizational structures the inspection team might take, but the governing factor is that the team as a group should be competent to successfully and efficiently conduct the safety inspection of the entire nuclear facility. A suggested team organization, one which lends itself to functional adaptation is listed below:

1. Team Chief.
2. Operations and Maintenance.
3. Instrumentation and Control.
4. Organization and Administration.
5. Health Physics.
6. Safety Inspector.

The areas of responsibility for each member of the above team are functionally apparent, with the exception of the team chief and the safety inspector. The function of the team chief will be more clearly brought out later, while the duties of the safety inspector can be more appropriately explained and discussed here. As each member of the team is concerned with an explicit operational area of the nuclear reactor facility, it would be beneficial to have one inspector who would be free to survey the entire facility, looking for anything that might endanger the safety of personnel or equipment. This leaves the team chief free to focus his attention on supervising and counseling the entire inspection team. (Ref 10)

<u>Pre-inspection Action.</u> When the decision has been made that a specific nuclear reactor facility is to be inspected, it is necessary to notify the command concerned and the individuals selected as members of the inspection team. The notification to the command might include information on what is to be inspected and any area or operation of particular interest, as well as an inquiry as to the convenience of the proposed date. This notification should be approximately 60 days in advance to allow preparation for the inspection by both the facility management and the inspectors. One might even encourage the facility to report on their own self-inspection results and their incidents. The inspection team should then conduct a pre-inspection review. This is to allow the team to become familiar with the facility and to determine the effect any modifications have on the safe operation of the facility. The scope of this review should include, but not be limited to a review of:

1. Hazards summary reports
2. "As-built" drawings
3. Manufacturer's literature
4. Periodic operating reports
5. Incident or malfunction reports
6. Design reports

7. Plant operating and maintenance publications and procedures
8. Emergency plan for control (Ref 11: 2)

When the inspection team, having completed the preinspection review, arrives at the facility, it is pertinent that certain coordination be affected by the team chief. The air command concerned should be visited by the team chief for purposes of effecting personal coordination. (Ref 47) Then the reactor facility commander or manager, along with key personnel of his choice, should be briefed as to the proposed conduct and schedule of the inspection. In order to effect more coordination and cooperation between the inspection party and the facility personnel, arrangements might be made to have a facility representative accompany each member of the inspection party throughout the inspection. Also, this allows a personal interchange, in a timely way, of the background of the inspectors, if it is considered appropriate. It should be pointed out however, that a lot of information can be obtained through casual conversation with the operators when there are no company representatives around. (Ref 47)

<u>Conduct of the inspection</u>. The inspection should be conducted over a two to three day period in order to allow ample time for a thorough inspection, but yet allow the facility to resume normal operations as soon as possible. If feasible, of course, the optimum situation would be to disrupt in no way the facility schedule, but this is of secondary importance. Each inspector, should know exactly what is to be inspected. For example the first step might normally be to have the reactor shutdown, so that startup procedures can be observed. The team chief should inspect the overall management of the facility from the viewpoint of safety, but he should make himself readily available to all team members desiring inter-

mediate conferences or advice. The success of the inspection is enhanced by such consultations, when needed, and all team members should be encouraged to see the team chief at anytime to discuss questionable matters. (Ref 9) Also, a team critique at the close of each day is effective in bringing out areas of mutual interest and concern. The team chief can, through such daily critiques, insure that the inspection is progressing properly. Basically, the team chief is concerned with the organizational structure of the facility and the effectiveness with which this structure furthers the facility safety program. Once the inspection is completed, the entire inspection party should have a full critique to discuss the results of the entire inspection. This is the time at which an inspector can request a counter-check or recheck by another inspector, if such is deemed advisable. At this time the team chief has the opportunity to make decisions, if necessary, on what shall be included or deleted from the inspection notes. Then the facility manager and facility representatives of his choice are informed of all findings of the inspection party, both good and bad. This is the time at which clarification of previous comments may be requested by the facility personnel. Such action may indeed shed further light on the cause of discrepancies, and most important, allows management to immediately correct what is mutually agreed to be a dangerous situation. Then carbon copies of all written comments made by all inspectors should be given to the facility manager. This enables the facility personnel to institute immediate corrective action where possible, and in fact should encourage such action. (Ref 9)

During the following period of approximately thirty days, the team chief supervises the preparation of the inspection report, which should consist of two parts:

1. The formal report, which includes only factual information (Ref 3: 10-11)
2. The letter of transmittal, which includes opinions and non-factual material such as comments on areas of outstanding performance. (Ref 11: 6)

In as much as this is a military inspection of an Air Force contractor facility in some cases, an inspector should guard against making recommendations or comments which may be used as improper substantiation for additional contract funds. This has happened on several occasions.

In order to insure maximum response and corrective action by the facility management, it is suggested that tradition be ignored in the distribution of the final inspection report. Specifically, the inspection report itself should be sent directly to the facility commander or manager, who in turn would complete all possible corrective action then endorse the report up through subsequent levels of command to Headquarters, USAF. Such a procedure is not presently used by the Air Force, which distributes the inspection report down through command levels to the facility. If the facility is found to be satisfactory, only a report of "satisfactory" would be sent to Headquarters, USAF. In case of an unsatisfactory rating, then a copy of the inspection report would be sent to Headquarters, USAF along with the "unsatisfactory" rating. Under any circumstances, the facility should be afforded the opportunity to initiate the endorsed report up through channels. It is felt that this procedure might also initiate more interest and support from the command levels between Headquarters, USAF and the facility manager. (Ref 9) Thus the primary purpose of the inspection, to insure the maintenance of adequate safety standards and conditions at USAF nuclear reactor facilities, is more apt to be achieved successfully.

IV. A Detailed Safety Inspection Guide Operations and Maintenance (Ref 27: 25-27, Ref 33)

Introductory Comments.

This inspection guide is intended to provide the inspection team with a basic reference which may be used to gain an insight into and an understanding of the requirements necessary for safe operation and maintenance of a nuclear reactor facility. As explained earlier, due to time limitations, this guide is restricted to the areas of operations and maintenance. The area of operations is considered to include the safety organization and management structure of the facility, its operations unit, and the operational safety aspects of the reactor. Other areas such as controls and instrumentation are included by necessity, for it is inconceivable that the operations unit or the operational safety of the reactor could properly be inspected without inclusion of all allied aspects of safety. Engineering and physics sections, as they apply to operational safety, are also considered applicable. These allied areas will also be inspected by other members of the team, and hopefully, in much more detail than is indicated in this checklist. Maintenance includes both periodic and preventive maintenance procedures as they pertain to safe operation of the reactor and its ancillary equipment.

Undoubtedly, this guide will not apply totally to every facility, for the guide is purposefully written in such a manner as to be generally applicable. The inspector should not allow himself to be restricted or channelized by any checklist, rather he should use it as a guide upon which to expand when necessary. The areas of operations and maintenance were chosen because of their overall importance in the field of facility safety, and because it appeared that they offered the most fertile reception of safety inspection.

The checklist was prepared with the use of many reference sources, as indicated in the bibliography, but the author used two sources predominantly. The general sequence of inspection is that used by the Atomic Energy Commission, as outlined by the March, 1955, issue of Nucleonics The acutal checklist questions and support were obtained from notes prepared by Major James R. Bohannon, Jr. of the Physics Department, Institute of Technology, Wright-Patterson Air Force Base, Ohio. These notes were used by Major Bohannon in the Air Force DNSR safety inspection of the Convair Nuclear Facility in March 1960.

The detailed safety inspection guide is divided into two areas, pre-inspection action and the inspection of the facility. The inspection of the facility section is broken down in such a manner as to provide continuity and logic for the reader as well as an inspector who might apply the checklist. The facility inspection consists of the following sections:

1. Discussion with operating supervisors.
2. Safety committee.
3. Reactor operating personnel.
4. Inspection of reactor operations sequence.
5. Controls and instrumentation.
6. Heat generation, transfer, and distribution.
7. Safety systems.
8. Shielding and radiation control.
9. Operational procedures and records.
10. Experimental systems.
11. Maintenance.

Pre-Inspection Action.

Prior to arriving at the facility, the inspectors should spend ample time in becoming thoroughly familiar with the reactor facility. The major portion of this task lies principally in a detailed study of the Hazards Summary Report and the approved operational "charters" issued to the facility by the ACRS and the Air Force. The various design features of the reactor and its ancillary equipment, as approved by the Atomic Energy Commission, should be carefully noted. Special attention should be devoted to the design parameters, control and instrumentation, and analysis of credible accidents. Finally, the inspector should study all available data on the organizational structure of the facility. This will enable the inspector to understand the makeup and functions of management, the safety committee, operations, and the health-physics group. A very close study of the standard operating procedures employed by the facility is of particular importance to the inspector. Only after thus preparing himself is the inspector ready to proceed with the inspection.

Inspection of the Facility - Operations and Maintenance.

Discussion with operating supervisors. "About half a day is spent discussing procedures, policies, and organization with the operating supervisors". (Ref 27: 26) These people are management personnel and are not directly connected with operating the reactor, but they are policy makers and as such directly affect the safety of operations. It is at this time that the inspector determines how well-defined the operations organization is as it actually exists and how effectively it functions in furthering safety at the facility. Also, the inspector can establish a feeling of confidence in the inspection team among the organization being inspected.

1. Do the management personnel of the facility periodically review the operations? (Ref 35: 1)
2. Have the responsible people read and made themselves cognizant with the Hazards Summary Report? This is a "must", and compliance should be 100 per cent. If personnel responsible for the operations of the facility do not understand the Hazards Summary Report, they can hardly establish thorough safety procedures. Are they aware of the technical specifications for the reactor?
3. Do the operations supervisors know what abnormal actions that the operators and experimentors cannot take without permission from the supervising staff? (Ref 27: 27)
4. Who makes decisions to countermand routine operating procedures? (Ref 27: 27) This authority should be invested only in a member of the supervising staff, and if a potential hazard is involved, in the safety committee. (Ref 3)
5. Is the reactor operating group separate from the experimental group? The full-time operating group should be concerned with the safety of the reactor, rather than with the completion of any specific program. (Ref 27: 26)
6. Are the responsibilities of the operations supervisors, operators, and other key safety personnel available in written form? (Ref 34) Are these realistic?
7. Is there a program of advance scheduling of operations and maintenance? This is necessary for operator planning and continuity of effort, both of which contribute to operational safety.

8. Have all theoretical aspects or predictions been verified by the operating history of the reactor? (Ref 37) If there have been discrepancies, they should have been reported and the Hazards Summary Report changed accordingly, if applicable.
9. Are critiques held on reviews of incidents at other facilities?
10. Do the supervisors avail themselves of literature on nuclear safety, such as the A.E.C. quarterly, Nuclear Safety?
11. Does management fully understand the potential hazards they have in their facility and the magnitude of potential losses?
12. Are there any areas of safety which are being jeopardized by Air Force "red tape"? If so these should be pointed out in the report.
13. Did the organization undertake a self-analysis program after being notified of the inspection? If so, are there any concrete results?

Safety Committee. If properly staffed and utilized, the safety committee can prevent accidents by preventing circumstances which increase the possibility of accidents.

1. Is there a safety committee with clearly defined responsibilities and organizational structure? "The responsibility of the safety committee shall include the review and evaluation of all aspects of operations involving nuclear safety, and advice to management, in writing, of its views and findings. The responsibilities and duties of the safety committee, and procedures for execution thereof, shall be set down in writing and approved by management of the reactor facility. (Ref 35: 1) The structure of the safety committee must be such that the committee is competent to rule on

proposed reactor modifications and so that every one is up-to-date on the response and characteristics of the particular reactor type. (Ref 37)

2. Specific duties of the safety committee should include, but not be limited to the following: (Ref 35: 1-2)

   a. Insure that the provisions of the charter are being complied with by the facility.

   b. Review and evaluation of proposed experiments.

   c. Periodic review and evaluation of operating procedures.

   d. Review and evaluation of any proposed modifications in the reactor or ancillary equipment affecting the safety of operation.

   e. Review and evaluation of local regulations for the handling of materials and equipment which involves radiation hazards.

   f. Review and evaluation of personnel radiation monitoring procedures.

   g. Approval or disapproval authority for the preceding procedures, regulations, and proposed actions.

   h. Investigation of accidents involving the reactor and/or its allied equipment and all other abnormal situations involving safety.

3. Is the safety committee periodically briefed on accidents occurring which might apply to their facility? (Ref 37)

4. Do the members of the safety committee avail themselves of periodicals on safety, such as Nuclear Safety? (Ref 37)

5. What is the membership of the safety committee? The safety committee should be composed of persons of maturity and

experience in the various specialties important to operational safety. Usually, depending on the type of facility, the committee should include a senior radiological safety officer or health physicist and a senior technical person not associated with the reactor project, but knowledgeable in reactor technology. (Ref 35: 2) The safety committee might also include engineers representing the internal operations group, management, reactor physics, and the experimental group. These people are intimately familiar with the day-to-day operations of the facility and can greatly add to the effectiveness of the Safety Committee.

6. Are the members of the safety committee aware of their responsibilities? (Ref 37)

7. Does the safety committee formally report all incidents to management and the DNSR, if applicable? (Ref 37) This brings up the question of who is responsible for indemnification, the contractor or the USAF. Find out if there is a clear understanding of liability. (Ref 37)

Reactor Operating Personnel. The purpose of this phase of the inspection is to determine if the reactor operating personnel, from the operations manager down through the reactor operators, are capable of performing their duties in such a manner as to enhance the safe operation of the reactor. In order to accomplish this goal, the inspector should first have a discussion with the operations manager. The reactor operators should be interviewed to ascertain their knowledge and understanding of the reactor and of the operating procedures. It should be kept in mind that there are differences in the policies followed regarding the responsibilities of operators, as well as the degree of knowledge pertinent to reactor phenomena. Some facilities only allow operators to push buttons and record information, others allow operators to perform preventive maintenance, and still others allow operators full leeway in reactor operations and maintenance. The inspector must be certain that he understands the delineation of operator responsibility before asking questions of the operators. Further the inspector should determine whether or not the operators understand this delineation. (Ref 37)

1. Does the operations manager know what his specific responsibilities are, and is he technically qualified to fulfill these responsibilities? The operations manager not only should have the requisite background and experience, but he should be abreast of his field. Is he cognizant of recent incidents and new data pertaining to his type of facility?
2. What kind of background check is made on potential operators as to their education, experience, and personal habits? Needless to say, it is vital that an operator not only be qualified technically, but that he be qualified character-wise also. (Ref 37)
3. Are the operators skilled technicians who are thoroughly familiar with their reactor? (Ref 27: 26) The operator should be familiar with, and have an understanding of, the following aspects of the reactor: (Ref 36: 7)
   a. The general design and operating characteristics.
   b. The control and safety mechanisms.
   c. The control-station instrumentation.
   d. All pertinent standard operating procedures.
   e. Emergency shutdown system and procedures.
   f. All other aspects of operational safety such as ability to

read and interpret the instrumentation and to know the purpose and functioning of radiation monitoring equipment directly applicable to reactor operations.

4. In order to ascertain whether the operators have the preceding qualifications, some questions to be asked of the operators are listed below: (Ref 37)

a. Does the operator understand the purpose of referring to the reactor log when first coming on duty? This is a very important point, for the operator should understand that vital information on parameters, instrumentation and other reactor history is available in the log and can influence safety features. Additional information on the contents of the log may be found under operational procedures and records, a later section of this guide.

b. Does the operator know the various scram mechanisms of the reactor?

c. Does he and can he ever override safety devices, with or without approval of designated supervision? This is the time to ascertain if the operator has the same understanding of this procedure as the supervisory staff.

d. Does the operator know who has the authority to restart after a scram? Find out under what conditions he may restart and ask him if a restart is the same as an initial start-up. (Ref 37)

e. How does he know that his chambers have continuity? What action does he take if he determines loss of continuity?

f. How would he check for a ruptured fuel element and what action would he take if he detected one? The importance of the operator recognizing this problem is pointed out in Appendix B in the analysis of the Sodium Reactor Experiment incident.

g. Does he know the maximum power levels and the minimum period at which the reactor can be safely operated? Who is authorized to change these levels, and is there an audio announcement of power changes by the operator?

h. Does he record all operational problems in the log and report pertinent information to his supervisor? The operator is the key link in the safety chain, for he is the individual who observes at firsthand all irregularities. If he fails to report all such items, corrective action cannot be taken.

i. Does he understand the function of the control rods, safety rods, shim rods, and regulating rods?

j. What would he do if the cooling system failed?

5. Is there a formal training program for the reactor operators? There should be such a program to assure operator familiarity with the plant. The program should include instruction, written examinations, and technical seminars on procedures, with discussions of other facilities and incidents. Active participation should be encouraged by management, and the program should aid in remedying operator deficiencies. (Ref 27: 26)

6. What is the relationship between the reactor operators and the experimental or technical group? (Ref 3: 2-10) Each group should understand their specific areas of responsibility and closely coordinate experiments, tests, and other non-routine

programs. This should be accomplished, following written procedures, through dry-runs and rehearsals. (Ref 37)

7. Do the operators and/or supervisors know the health physics limits on exposure from the various sources available?
8. Does the operator understand the policy on maintenance during operations?

Inspection of Reactor Operations Sequence. The inspection during the performance of the operating tests is the key phase. Startup on the daily standard procedure is made and then the reactor is operated at various power levels. Then the reactor is shutdown on a power level scram to test the actual operation of the power-level trip mechanism. The reactor is restarted after the short-term shutdown and then made to scram on the period trip. Finally the reactor is restarted and shutdown by standard procedures.

Additional steps are made to test and inspect as many safety circuits as possible during the actual operation of the reactor. These circuits would include detectors, rod action, other scram devices, and all other primary safety features. Completion of the reactor operations phase of the test might take anywhere from four hours to one day or longer. These tests must be conducted as thoroughly as possible, but with the least possible disruption of the operating routine. (Ref 27: 26) Below are typical items or questions for the inspector to bear in mind during this phase of the inspection.

1. Look for the following desirable features on the control panel. (Ref 12: 82)
   a. Miniaturized panel components.
   b. Natural "expected" movements for adjustments.
   c. Switches mounted below meters or indicator lights.
   d. Spring-loaded or fail-safe switches.

e. Legible nameplates.

f. Red marks on meter faces to indicate potentially dangerous position.

g. Status of valves and pumps indicated logically with colored lights (e.g. green to indicate normal situation).

h. Panel size appropriate for average-sized operator.

i. Controls not so automatic that the operator might fail to interpret the significance of an unusual situation.

"To decrease the human errors that cause most of the unintentional plant shutdowns during reactor operation, control-panel layouts should simplify the operator's job by providing information in a more meaningful form". (Ref 12: 82) All of these features may not be in the console, and if not, appropriate comments might be made in the inspection report, so that if future modifications are made in the console, such features could be incorporated into the changes.

2. How thoroughly does the operator check the reactor before start-up?

a. Does he refer to the operating log to review past performance and history of the reactor from the last time he was on the console? As previously pointed out, the operator must be aware of changes in the reactor parameters and any recent operating problems. For example, he must predict the rod positions for criticality.

b. Does he follow a detailed checklist? This should be mandatory for even the most experienced operators. Humans will err, and use of a written checklist will avoid many errors (see Appendix B which analyzes the HTRE-3 incident, in which an erroneous voltage setting contributed to a power excursion).

c. Does he check the "outside" of the reactor system for such items as open valves, loose cables, and control rods positioning?

d. How does he test all of the scram circuits and monitoring equipment? It is very important that all operators should use the same equipment for calibrating and testing circuits, as different results will be obtained using different equipment. The best solution is to tag the calibration equipment and store it when it is not in use. (Ref 43)

e. Does the operator check control rod movement and position indicator? (See the NRX incident in Appendix B).

f. Does he check all recorders to ascertain that all are writing properly?

g. Does he check the console to make sure that all switches are in their proper positions?

h. Does he physically check the reactor if it is an open and easily accessible type?

3. Is there sufficient space for the operators to perform properly? In addition the noise level and distractions should be checked here. There should be limited and controlled access to the area.

4. Is the operator aware of void values, if applicable, and such specialized problems as gamma heating? (Ref 37)

5. Is the source strength sufficient for startup?

6. Does the startup go smoothly?

a. Does the operator follow a detailed checklist?

b. Does he have too many things to do at once?

c. Does he monitor pertinent instruments? He should know exactly where the reactor is, relatively, and should be able to immediately detect any abnormal situation which warrants operator action. Ignoring information of this nature, or not understanding it, can lead to unsafe conditions (see Appendix B for the analysis of HTRE-3 incident, where a much shorter period than was predicted was indicated and no operator action was taken).

d. Are subordinate checklists used by those personnel checking in with the operator? Some inspections have revealed these personnel acting by memory or habit. That can be dangerous! (Ref 37)

7. Does the reactor operate at the various power levels in a stable manner, or does it require constant attention and control by the operator? Check to see if the circuits are noisy and if there have been an undue number of accidental scrams. What action did operators take to remedy this situation?

8. Does the reactor change power easily, or does it respond slowly and sluggishly to commands from the operator?

9. Is it possible and/or necessary for the operator to make awkward range or detector changes during the change in power?

10. Does the reactor scram on excess power at the prescribed level?

11. Does the operator check everything according to the routine checklist before startup after the short-term shutdown?

12. Does the reactor scram at the prescribed period?

13. If possible, test the remaining safety circuits during operation

(e.g. the detectors for stack-gas radioactivity, rod action, and the temperature and low coolant flow scrams).

14. What is the temperature coefficient of reactivity? This is sometimes difficult to obtain, but it is important information to have, as a positive temperature coefficient requires additional safety devices.

15. Does the reactor operator have a shutdown sequence list which requires use of all scram mechanisms? This should be a regular program.

16. Were there at least an operator and a shift supervisor on duty in the reactor area at all times? This should be a minimum requirement and should be in writing. (Ref 35: 3)

Controls and Instrumentation. The controls and instrumentation are the means by which the operator is informed of the status of the reactor and is able to influence the reactor. The inspector should determine if the controls and instrumentation have adequately functioned in the past (and still do) and if they are and have been properly employed. It should be determined if any modifications have been made that are not reflected in the Hazards Summary Report. Such changes should be closely investigated for their effect on operational safety. These changes should have been coordinated with the reactor physics, engineering, and safety committee groups. (Ref 3) Too, he should inspect the records to see what information is reflected on the performance of the controls and instrumentation. Some of the points the inspector checks in this area, as well as safety systems and others, are really design problems. However, due to changing design philosophy and changes in the facility program, what might have been considered best at the time operations was initiated may no longer be best. For this reason, the inspector is encouraged to pass on to the facility personnel the benefit of his experience by appropriately commenting when advisable. This will allow recommended modifications and improvements to be incorporated into any such contemplated program planned by the facility.

1. What is the worth of each rod and all possible rod combinations? Depending on the reactor design, no single rod should have more than about four per cent delta K. (Ref 29)
2. What is the program for placement of rods in the core, and how is this controlled operationally? Are the rod values the same for all loadings? (Ref 37)
3. What are the possible rod combinations that will allow shutdown? (see the NRX incident in Appendix B).
4. Would a malfunction allow a safety rod to drop through the reactor core? (Ref 28: 21)
5. Could a safety or control rod inadvertently withdraw a fuel element? (Ref 28: 21)
6. Is the rod-drive mechanism irreversible? It should be so that forces on the rod cannot move the drive mechanism. (Ref 30: 49)
7. Are the control rods capable of withdrawal at a rate greater than the design speed? (Ref 32: 80)
8. Is the rod-withdrawal system of the motor-driven-cable type? This older system is not conducive to safety, as it increases the possibility of accidental withdrawal by overhead cranes and other equipment. (Ref 31: 10) The BSR at Oak Ridge, for example, replaced the old stringer-type system it formerly used.
9. Are the withdrawal timers fail-safe? (Ref 43)
10. If the rod-drive system is electrical or mechanical, is there

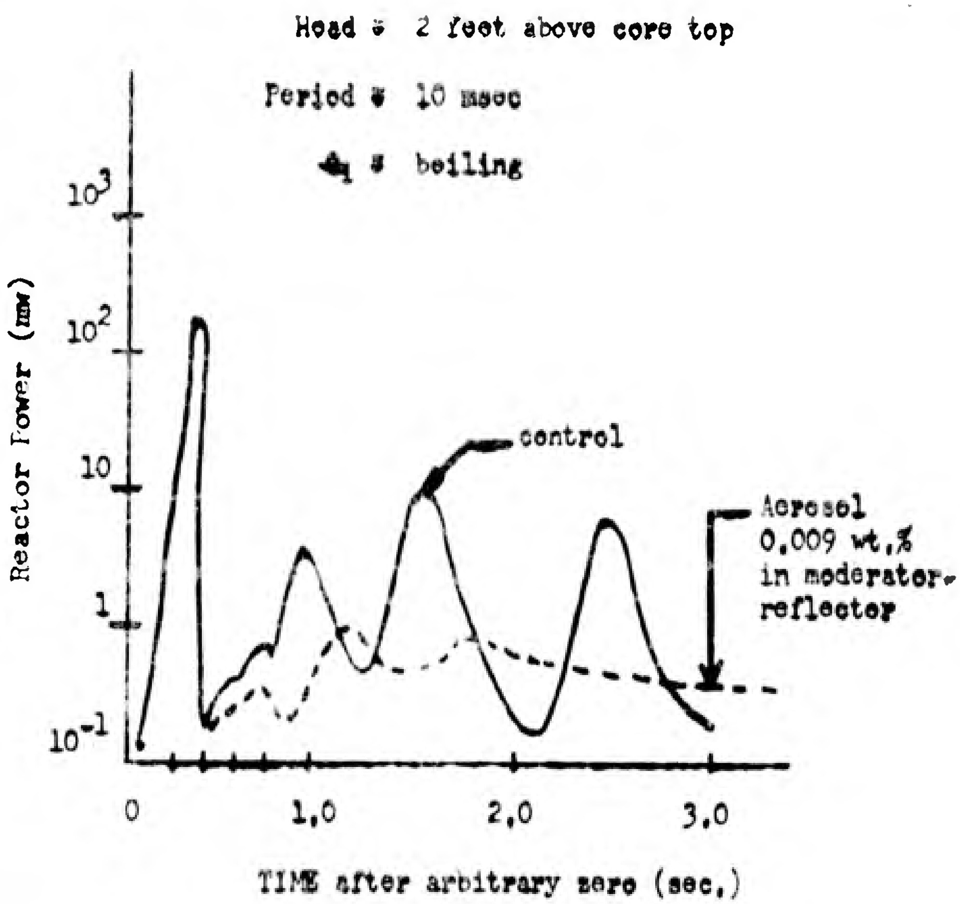

Fig. 2

Effect of Detergent on Power Burst Shape

(from Ref 44: 9)

a gravitational backup system?

11. Can the automatic control stay in the "on" or demand position after an automatic scram and restart the reactor? (Ref 37)

12. Attention should be directed to the following: (Ref 29)

    a. Mechanical adequacy of rod structure.

    b. Thermal stresses and distortions of rods.

    c. Corrosion or solubility of rod components.

    d. Buoyancy or flow effects of coolant or moderator.

    e. Nuclear burning of rod poison.

    f. Radiation damage in rod materials.

13. Are additional methods of control, such as a movable reflector, employed? Is this method used in conjunction with control rods and is reactivity change per time and position accurately known? Tests were performed with SPERT I to determine if improving the heat transfer properties of the moderator might not cause the reactor to shutdown more quickly. This possibility was investigated by adding a wetting agent, Aerosol-OT, to the moderator. The results indicated no drastic effects, as shown in Figure 2 on the facing page. (Ref 44: 9)

14. Are performance checks of components and systems scheduled, and are check methods outlined? Are the results recorded?

15. Is the clearance between rods and fuel, etc., such that there is a possibility for misalignment, jamming, or hang-up? Design clearances may not have taken into account rod swelling or corrosion effects.

16. Is there a fail-safe or other safety feature to back up the control rods? In addition, all withdrawal requests given by

the operator should be subjected to being over-ridden by an automatic insertion mechanism if necessary. (Ref 43) The reader is referred to the HTRE-3 accident in Appendix B for an example of the danger here.

17. What is the nature, strength, location, and distribution of the neutron sources?

18. Is the power level of the reactor continuously indicated during operation and during shutdown while manipulations on the reactor are in progress? (Ref 29)

19. Are the detectors and associated instruments, equipped with automatic level and rate-of-rise trips, capable of responding to the neutron flux in the start-up range? (Ref 29)

20. Are there at least two, preferably three, independent flux monitoring channels? (Ref 29)

21. Are both power and period scram protection constantly employed when the reactor is operating below one megawatt? (Ref 29)

22. Are test signals introduced into the equipment to check for proper operation? (Ref 32: 79)

23. Are coolant flow and temperature and center-line temperatures adequately measured?

24. What provisions are there for malfunction?

25. Are the time constants consistent with the phenomena monitored?

26. What spare units are available for control systems and are they checked out before use? The location of spare units and certification of their checkout (to include method used) should be recorded. (Ref 37)

27. Can the controls be manually removed from the reactor once the

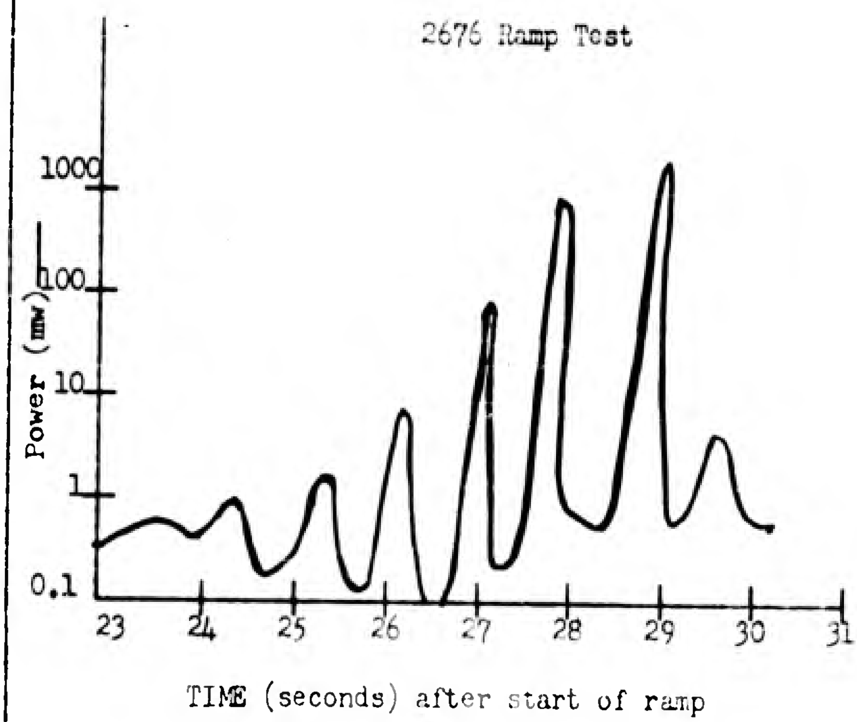

Fig. 3

Power Divergence for Boiling Ramp Test
SPERT I

core is loaded?

28. Is supplemental instrumentation used in case of new core configuration studies and analysis?

Heat Generation, Transfer, and Distribution. Are the thermal aspects of the reactor complex handled in such a manner that potential hazards are recognized and minimized? The inspector should be primarily interested here in ascertaining that the reactor operating history confirms the predictions made in the Hazards Summary Report. The following items should be checked by the inspector for confirmation: (Ref 3)

1. Heat generation rates in the core, blanket, shields, components, and structures.
2. Normal operating temperatures.
3. Distribution of fuel and maximum specific power.
4. Effect of operating temperatures on stress, expansion, thermal shock, fatigue, and corrosion.
5. Limiting powers and temperatures. An interesting example of what to expect under boiling conditions is exemplified by one of the SPERT I tests. This boiling ramp test, with 1.9 per cent delta K added at the rate of 0.07 per cent per second, indicates the extreme divergence in the reactor power. (see Figure 3, facing this page). (Ref 45: 8-9)

In addition, the inspector should look into several other key areas which are very important to safe operations.

1. Can vapor voids form in the primary. If this is possible, determine how they are prevented from entering the core and what the consequences are, if they do enter the core. (Ref 37)

2. What type of emergency cooling system is provided? If there is none, then there must be either inherent safety or system configuration to protect the reactor. For example, multiple coolant loops might be provided, as a reactor with four coolant loops is inherently safer than a reactor with only one loop. "Reliance on such a multiplicity of loops and components may be misleading, however, since a single rupture inside the block valves to the loops could result in the loss of coolant to the entire system". (Ref 14: 40)

Safety Systems. The safety systems are those devices and components which act as coupling agents to restrain, restrict, or activate any action that is pertinent to the safety of the reactor system. The inspector should determine if the safety systems are adequate, functioning properly, and are properly employed. In this area the inspector is concerned with determining if the safety system still consists of the same components, circuitry, and operational methodology as indicated by the Hazards Summary Report. Too, it may be that changes in design philosophy or facility programs dictate additional less, or modified safety system components or employment. The following points are itemized for the inspector to check, keeping the Hazards Report and modifications in mind. (Ref 3)

1. What is the connection system between instruments and control and/or safety rod actuators?
2. What restrictions exist on rod motion during startup and normal operation? Are these restrictions electrical, mechanical, or procedural? How, when, and by whom are the restrictions imposed and removed?
3. What process variables can cause automatic safety action or

annunciation? These might include neutron and gamma flux, temperature, pressure, liquid level, and radiation level. Several of these are desirable, but the employment of unnecessary variables to effect safety action can cause operators to form the habit of overriding safety systems. Is this the case here?

4. Is each variable in the flow system backed up by one or more independent systems? For example, is a temperature device backed up by a flow device? How many independent devices monitor the same variable?

5. Is there more than one safety setting for some variables? Particular attention should be given to their basis for choice, method of change, action to change, and who makes the choice. In addition, the inspector should determine the standards used in checking these changes. The hazard involved in improper methodology here is exemplified by the NRX accident analysis in Appendix B, where the cooling system capability was reduced without adequate operating experience to determine the correct reduction.

6. Is there more than one safety action? Examples of this are rod reversal and slow shutdown. What is the basis for action and what variables are monitored?

7. How and by whom are trip levels for safety action chosen? Do they change? This is an extremely sensitive area, and only qualified reactor physics and electronics personnel should exercise responsibility here. (Ref 37)

8. What minimum portions of the safety system are required for

operation of the reactor at various power levels? This again is a highly sensitive area, and such dicisions should be clearly defined in writing, dated, and approved by the safety committee. (Ref 37)

9. What malfunctions of the safety system components cause safety action or give a signal of malfunction. An inquiry is proper here to determine whether or not the operators have become accustomed to ignoring malfunction signals such as burned out lights, etc. Are annunciators foolproof?

10. What are the time constants of the various safety systems, such as relay, delay, and rod drop? Do they appear to be consistent with the monitored phenomena and have they varied since operation began?

11. What by-pass features are incorporated into the safety system? How are they controlled operationally? Look into this closely to make sure that authority for use of these by-pass features is properly vested and is approved in writing by the safety committee. Is the system foolproof? (Ref 37)

12. What interlocks are incorporated into the safety system and how are these controlled operationally? What is the interlock-bypass design philosophy, and how does operations define interlocks and bypasses? This is a very sensitive area and should be looked into closely to determine whether or not operations has formulated a "loose" philosophy on definitions and instructions. The philosophy should be clearly spelled out, in writing, by management and approved by the safety committee. It should be such as to disallow the bypass of any interlock which would create a

dangerous situation; under any circumstances, keys or other positive means should be required for activating by-passes.

13. What spare units are on hand for replacing safety system components? Are these spare units tested, using a standard written checklist, before being placed in use? (Ref 37)

14. What supplemental safety action is available? Examples of such provisions are liquid level, temperature, and poison. How are they initiated, what is their effect, and how quickly do they act?

15. What are the power supplies for the safety, control and instrumentation systems? What backup is available in case of failure of the primary or purchased power supply? Is the power supply independent from external phenomena? Poor voltage supplies can cause electrical noise and hinder the proper functioning of the control, instrumentation, and safety systems? (Ref 37)

16. What is the schedule and method for checks on the performance of components and systems? What are the results of past tests? These checks should be made both by test signals and during actual reactor operation. (Ref 37)

17. If coincidence circuits are used, are they really independent? Has an analysis of failure rate been done, and if so, by whom and by what method?

18. What parts of the system are or are not fail-safe and how do they define fail-safe?

19. What system is provided for the surveillance, calibration, and checking of the safety system by the operating crew? This should be clearly outlined in writing and inserted in the

reactor log.

20. Is there at least one level scram (neutron, gamma, temperature) set to ride down in the near vicinity of actual level at any time during start-up or operation at levels substantially below nominal maximum power? Is the level properly advanced as necessary as power increases? (Ref 29)

21. Are there interlocks in all flux detectors which will cause reactor scram if the high voltage supply to the chamber deviates substantially from the proper valve? (Ref 29)

22. After maintenance or alteration of a safety channel, is a complete recheck of response made, to include interlock activation by an appropriate signal? (Ref 29)

23. Are more interlocks provided than are needed? It is essential that the necessary interlocks be provided, but unnecessary interlocks can be hazardous by encouraging operator deactivation. (Ref 29)

24. If there has been an approved power rise, will the reactor emergency and shutdown cooling system still suffice to meet the safety requirements of a maximum credible accident? (Ref 14: 38-39)

Shielding and Radiation Control. The inspector is only interested in shielding and radiation control from the viewpoint of operations and maintenance. He should ascertain what periodic tests are conducted to verify the effectiveness of the radiation control program and equipment. The schedule and program for radiation surveys should be examined, along with the records of past surveys. Too, the inspector might inquire into the following specific items: (Ref 38: 6-7)

1. How are personnel controlled in restricted areas?

2. What is the attitude of the personnel toward radiation safety?

3. Is the special work permit procedure fully outlined in writing, and is it understood by all personnel? Do they comply?

4. How much fuel is stored on-site? (Ref 37)

5. What kind of radioactive waste is there and what is the authority to generate isotopes? (Ref 37)

6. Are there visual or audible alarms in areas which might inadvertently be contaminated by a radiation source?

7. Is it possible to inadvertently raise fuel elements to the surface of water shields, or expose personnel to fuel in any manner?

Operational Procedures and Records. Routine operation of a reactor facility is very difficult to obtain, and for a general-purpose research reactor, it may be impossible. It is highly desirable, however, to standardize as much of the operation as is possible. (Ref 40: 62) This can be achieved best through formulation of detailed operating procedures and maintenance of complete, accurate records. "The operating procedure should be carefully worked out and rigidly adhered to and even minor deviations should not be tolerated without review by more than one thoroughly competent person." (Ref 40: 16) Finally, a careful study of the record system must be made to see that it operates as an asset to the facility. The necessary records must be maintained, but record-keeping should not be pyramided to the point where it is hindering the safety of the operation or becomes busy-work. (Ref 41: 31) Briefly, the task of the inspector is to determine what procedures and records exist, their validity, and finally, if they are properly used:

1. Is there a properly composed log book with instructions on how it is to be maintained? The operator should know what information to enter into the log and when to enter the information. The log should provide a complete history of the reactor. Items that should be entered into the log are exemplified by the following:
   a. All scrams, to include type, cause, date and time, and corrective action, if any.
   b. All modifications to the reactor system, to include changes in parameters caused by modifications.
   c. All startups and shutdowns of a normal and abnormal nature.
   d. Rod calibrations and other parameter values.
   e. Non-routine testing of instrumentation and safety circuits.
   f. All maintenance work performed during shutdown and during operation.
   g. All unusual occurrences to include their cause, if known, and corrective action, if any.
   h. Pertinent instrumentation readings, as directed, on a scheduled basis.
   i. All experimental systems inserted into the reactor.
2. Are complete checklists provided in the log?
   a. Is there a written startup checklist? This should include pre-start checks also.
   b. Is there an interim checklist for use before startup is attempted after interrupted operation? This list will probably be brief, but it should include such items as checking to be sure that the low-level power instrumentation has been reactivated. "A temporary shutdown is

invariably accompanied by a certain amount of confusion, and it is at such a time that stardardization of sequence is invaluable." (Ref 41: 62)

c. How do the operating procedures guard against mistakes in instrumentation calibration, mistakes in setting of trip points of safety systems, and deviations from procedure? (Ref 3: 8-9) The HTRE-3 accident at the National Reactor Test Station in Idaho exemplifies the fact that these type of errors can occur. (See Appendix B)

d. Is there a shutdown checklist which insures that the reactor is secured and that equipment and instrumentation that have a relatively short life time are not left in operation. (Ref 40: 62)

e. Is there a checklist for normal operation which specifies in detail what the operator is to do and when he is to do it? (Ref 35: 72)

f. Are all of these checklists approved by responsible supervision? (Ref 35: 72)

3. Are all logs and records approved by responsible supervision? (Ref 35: 72)

4. Are there written procedures, approved by local management, for the immediate protection of the health and safety of personnel and prevention of damage to equipment under reasonably foreseeable abnormal conditions, moderate emergencies, and radiation emergencies? (Ref 35: 72) The Windscale incident (Appendix B) illustrates what can happen if prior plans for such a situation

are not made. The procedures should specify in detail what action is to be taken by whom.

5. Is there a detailed written decontamination plan? This plan should cover all aspects of all predetermined possible situations that might arise at the facility. The plan should not only specify methods, equipment, and names of personnel, but it should also specify the authority required to initiate the decontamination plan. This authority should, of course, rest with the health-physics group.

6. Are periodic drills directed, in writing, by management to check out planned emergency procedures? (Ref 35: 72) There should be critiques of the drills and written reports rendered to the safety committee.

7. Are there written procedures for handling visitors? Visitors should be accompanied by a member of the operating organization at all times. In addition, all visitors should be registered and provided with appropriate personnel monitoring equipment. (Ref 35: 73)

8. Does access to the control room, reactor area, and other areas around irradiation facilities require authorization of the operations superintendent or his designee? (Ref 35: 73) This should be stipulated in writing.

9. What disaster plans exist? These plans should be in written form, approved by the safety committee and management, and should be reviewed periodically. In addition, such plans should be integrated into other A.E.C., Air Force, and civil plans. Determine if facility personnel are cognizant of the

disaster plan. (Ref 3: 9)

10. Have other appropriate agencies been notified of the disaster plans? (Ref 3: 9)

11. Are faulty procedures detected by the operators brought to the attention of management for correction? (Ref 13: 39)

12. Are all procedures periodically reviewed by management and/or the safety committee to make sure that the procedures are accomplishing their purpose? (Ref 13: 40)

13. Do operators formally and periodically report to the safety committee and to management?

Experimental Systems. Operating experimental systems out-of-pile is hazardous enough, but operation in-pile is potentially even more hazardous. This is due to the possible interaction between the experimental system and the reactor, with such a possibility compounding the hazards inherent to either the reactor or the experiment alone. (Ref 42: 106) Inasmuch as many Air Force reactor facilities do operate experimental systems, this particular feature of operations is worthy of emphasis.

1. How are experiments planned and approved? Experiments should be planned by the responsible group and then reviewed by the administrative staff. The decision to proceed with or cancel the experiment should be made by a designated individual or the safety committee as a group. (Ref 1: 149) Operations should conduct "dry runs" or construct models, prior to initiating the experiment.

2. Is each experiment analyzed to ensure safety under normal operating conditions and all credible abnormal circumstances? "Four hazards must be examined, their probabilities assessed,

and their potential consequences determined:

a. Excessive temperature.

b. Material and component failure.

c. Sudden reactivity changes.

d. Chemical explosion.

These conditions are often interdependent". (Ref 42: 106)

3. How are temperature hazards analyzed? The basis for the analysis should be that the normal operating temperatures for any experiment should not exceed the limits of any component of experimental equipment or the reactor. (Ref 42: 117)

4. How are possible material and component failure analyzed? The following areas should be investigated by the experimental planning group:

   a. Maximum stresses.

   b. Pressure increases.

   c. Possibility of state changes.

   d. Effects from corrosion, erosion, and mass transfer.

   e. Irradiation effects. (Ref 42: 118)

   f. Potentiality of generating radioactive gaseous products. (Ref 37)

5. How are potential reactivity changes (due to insertion of an experiment) calculated? These should be precisely calculated and the fuel should be reloaded if necessary. (Ref 42: 118)

6. How are possible chemical explosions analyzed? The planning group should thoroughly investigate the compatibility of reactants, the extent of intermix, relative amounts of each material, and the various temperatures expected. In addition,

if the possibility of explosion cannot be completely eliminated, the planning group should closely evaluate the effects of an explosion on the container confining the reactants. (Ref 42: 118)

7. Are all irradiated materials from experiments closely accounted for by detailed and accurate records? (Ref 1: 150)

Maintenance. "The maintenance problem on a reactor is different from that of any ordinary industrial process only in that the additional hazards of reactivity and radiation are present". (Ref 41: 30) These differences are, however, extremely significant. The relation of proper maintenance of the equipment of a reactor, and in particular of the reactor itself, to the safety of the facility and associated people is extremely important. For this reason, preventive maintenance should be strongly emphasized, as potential hazards and costs must be minimized. The ultimate in design and operational procedures will soon be neutralized unless a comprehensive program exists for periodic and continuous maintenance. The degree of safety with which a nuclear reactor facility is being operated can be determined only if the inspection includes a thorough analysis of maintenance procedures employed at the facility. The following areas should be investigated by the inspector, as well as any others that are brought to the attention of the inspector.

1. Are appropriate written maintenance procedures provided by responsible management, and are these procedures being properly adhered to in the conduct of maintenance? (Ref 43)
   a. Are these procedures prepared by technical personnel?
   b. Are these procedures periodically reviewed by the appropriate technical personnel, and are revisions made where applicable? (Ref 37)

c. Are these procedures approved by the safety committee? (Ref 37)

2. Who actually performs maintenance? It may be performed only by operators, by operators and maintenance personnel, or only by maintenance personnel. The best method depends on the type of facility and the qualifications of the personnel. (Ref 37)

3. What are the qualifications of the maintenance personnel? Are these people chosen through an examination system and do they participate in a formal training program? (Ref 43)

4. Is all maintenance to be performed approved by the shift supervisor? (Ref 43)

5. Is only non-critical maintenance work performed during operation? (Ref 43) This should be explicitly explained in writing, so that everyone understands what is meant by non-critical.

6. Is all maintenance and repair that is performed inside the reactor vessel done only under the direct supervision of a responsible individual? (Ref 43) Are the names of such individuals included in a written directive covering this?

7. Are only the minimum number of personnel allowed near the reactor when maintenance inside the core is being accomplished? (Ref 43)

8. Is there strict recognition of the need for breathing masks, protective clothing, radiation exposure records and limitations due to radiation? (Ref 43)

9. Are the equipment and systems maintenance records accurately maintained so that they facilitate trouble shooting and expedite completion of all maintenance? These should also include equipment history and overhaul reports. (Ref 43)

10. Are all unusual or unexpected incidents that resulted or could

have resulted in serious injury or damage to equipment promptly reported to supervision? The following actions should be standard written procedures to prevent recurrence of such incidents encountered in maintenance work. (Ref 43)

a. Thorough investigation of all incidents.

b. Written reports, in detail, of all such incidents.

c. All operations should be suspended if there is any doubt of the safety in continuation of work (until the trouble is resolved).

d. Rules should be established and enforced to prevent the recurrence of a similar incident.

11. What are the qualifications of the personnel performing maintenance? This depends on the background and experience of the personnel as well as the facility training program.

12. Are those pieces of equipment likely to require frequent maintenance kept out of the neutron field? (Ref 15: 276) Although this is basically a design problem, it may warrant recommendation of modifications. Inquire as to how frequently this problem has arisen.

13. Does the reactor log indicate proper operator maintenance and does the log provide space for certification of performance of this maintenance. (Ref 35: 72)

14. Are radiation levels encountered in maintenance activities determined prior to initiation of work? (Ref 35: 72)

15. Is the safe-work permit mandatory before maintenance personnel undertake maintenance on the reactor or any of its components? (Ref 15: 276)

16. Does the safe-work permit assure that the level of radiation to which maintenance personnel are exposed is within permissible time limits for the time required to do the job? (Ref 15: 276)

17. Is there a separate "warm shop" where inoperable equipment that is only slightly radioactive may be repaired? (Ref 15: 276)

18. Is there a "hot shop" where highly radioactive material may be repaired by remote control? (Ref 15: 276)

19. Is defective equipment stored in a sufficiently shielded location or sent to the "burial ground"? (Ref 15: 276)

20. Are there specific written procedures covering new and preventive maintenance to be effected after shutdown and for reloading the reactor? (Ref 37)

21. What are the procedures (in writing) for maintenance of hardware, such as pipes, valves, generators, emergency power supply, etc? (Ref 37)

22. What is the formal periodic maintenance schedule for control and instrumentation systems and safety systems? (Ref 37) These should include calibration and testing procedures.

23. Are there records reflecting the completion of such maintenance as well as preventive maintenance? (Ref 34)

24. What modifications have the maintenance men performed? (Ref 37) Determine their understanding of who must give approval for modifications and ask them who checks their work?

25. Do the maintenance personnel think there should be any modifications to their maintenance procedures? Check to see if the supervising staff is aware of these suggested changes, if there are such. (Ref 37)

26. Are there irregularities of safety equipment which have been self-rectifying or unexplained that are still in the console complex? (Ref 37)

27. Are there areas where maintenance personnel could be inadvertently missed during an unannounced startup? (Ref 37)

28. Do drawings and schematics reflect the as-built conditions of components (electrical, electronic, and mechanical)? Have there been any changes to these and were they approved by the proper authority? (Ref 37)

29. Observe the physical loading or unloading of the reactor. This is a very hazardous operation, potentially, for errors in this sort of procedure are much more likely than errors in control-panel operation. This is particularly true of fuel elements being put into, or removed from, storage. (Ref 37)

V. Summary and Recommendations

The preceding section, a detailed guide for a safety inspection of the operations and maintenance areas of a nuclear reactor facility, keynotes the author's efforts to propose a safety inspection guide for use at United States Air Force reactor facilities. The basic aim of this guide is to further the efforts to standardize the methods and philosophies of inspecting these facilities. The author does not feel that the abrupt submission of a checklist for safety inspections would attain this aim.

In accordance with this belief, additional effort was directed toward providing supplementary material on reactor facility safety and facility inspection. Part of this effort was devoted to emphasizing the vital necessity of the safety inspection of reactor facilities. Doctor Edward Teller, one of the leading nuclear physicists in the world, commented that even with all possible inherent safeguards, no reactor is foolproof. (Ref 1: 135-157) Thus the potential hazard exists and its realization depends largely on the men who operate the reactor. (Ref 2: 27) The only manner in which the Air Force has of knowing whether or not USAF reactor facilities are safely operated is by inspection of these facilities. Then, too, it follows that an inspection by qualified, disinterested personnel will further safety programs.

In order to further provide an insight into the conduction of safety inspections, the philosophy of inspection was discussed. It was pointed out that safety inspections of reactor facilities are logical successors to the inspections conducted in the past. The basic goals are the same, that is to gain knowledge, attain conformance, and improve standards. Then the methods of conducting an inspection and the importance of the human element in reactor safety were discussed. Emphasis was placed on practical measures intended to gain the confidence of the facility personnel being inspected and on the importance of the attitude and competence of the reactor operations personnel.

Current inspection procedures were enumerated in some detail for two purposes. The first was to illustrate the regulatory actions preceding the safety inspection discussed in this guide, prior to operation and during the initial year of operation of the reactor facility. This was done by outlining the inspection procedures used by the Atomic Energy Commission and the self-inspection programs of industry and contractors. Secondly, a vast amount of experience has evolved from the past fifteen years of A.E.C. operation and inspections, and the author felt that the Air Force could benefit from this source.

Proposed inspection procedures for the Air Force were dealt with at some length. This included a background discussion of the organization and functions of the Directorate of Nuclear Safety Research which is the action agency for the USAF nuclear facility inspection program. Then the details of a proposed inspection system were given in conjunction with the reasons for the choice of the various aspects of the system. A suggested inspection team organization was put forth. Administrative details of conducting an inspection were proposed, including pre-inspection action to be taken, the inspection itself, and the reports rendered following the inspection.

This led logically into the detailed guide for a safety inspection of the operation and maintenance of a nuclear reactor facility. Some difficulty was encountered in attempting to obtain a detailed guide that would apply to any reactor type, but the guide should provide an insight into inspecting a facility. The areas of operations and maintenance were selected for emphasis because they appeared to be of major significance in facility safety, and they offered the most fruitful results from the viewpoint of an inspector.

Lest the reader should wonder at the seemingly obviousness of some of the checklist questions in the detailed guide, the author respectfully directs the attention of the reader to Appendix B, which is an analysis of past nuclear reactor accidents. For the reader who wishes to see an inspection guide pertaining to all aspects of a nuclear reactor facility, an abbreviated safety inspection guide is provided as Appendix A.

Nuclear reactor facilities have been inspected for safety for over fifteen years, and yet somehow, no real standardization of inspection procedures has evolved. This is partly justifiable on the grounds that almost ever reactor was of a new type for many years. Now, however, the A.E.C. and the Air Force are interested in attaining standardization where possible. Having spent a great deal of time in researching the vast amount of material available in this field, including some information which is as yet unpublished, the author makes the following recommendation. It is respectfully recommended that the Air Force consider the proposals made in this guide for use in effecting the USAF safety inspection program for Air Force reactor facilities.

Obviously, this inspection guide is not the ultimate in nuclear facility inspection. It is hoped, however, that it might prove to be thought provoking, both to the DNSR for administration of its safety inspection program, and to the inspector who seeks to determine if a facility is complying with established safety standards, and whether or not complacency has reared its ugly face.

Bibliography

1. McCullough, C. Rogers. Safety Aspects of Nuclear Reactors. Princeton, New York: D. Van Nostrand Company, Inc., 1957.

2. Mann, Marvin. "Roundup". Nucleonics, 17: 6 (June 1959).

3. Morris, Peter A. Reactor Inspection Guide (draft), 24 October 1959 (U. S. Atomic Energy Commission).

4. Schultz, M. A. "Comments on Organization and Procedure-Minimum Standards for Operation of Research and Test Reactors". From a paper presented in standards work at a special session of the American Nuclear Society in Cleveland, Ohio in April 1959.

5. Personal communications with Francis X. Gavigan, member of the Chicago Operations Office, A.E.C., Argonne, Illinois on 9 September 1960.

6. Report on Safety Survey of USAF Reactor Facility, 2-5 May 1960. (Georgia Nuclear Laboratory, Dawsonville, Georgia). Prepared by the Directorate of Nuclear Safety Research, USAF, Kirtland AFB, NM.

7. Internal Safety Aspects of Nuclear Activities at Convair, Fort Worth, Texas. 1 February 1960. FZM-1919.

8. Nuclear Reactor System, Safety Studies and Safety Rules. Air Force Regulation 122-15, Department of the Air Force. 15 December 1960.

9. Personal communication with Lt. Colonel Rudolph W. Staffa, Nuclear Power Division, OCE, Washington 25, D.C. (20 December 1960).

10. Personal communication with Major William C. Burns, Nuclear Power Division, OCE, Washington 25, D.C. (20 December 1960).

11. Guidelines for Technical and Health Physics Inspection of Nuclear Reactors. Department of the Army. (TB ENG 184) 4 August 1960.

12. Dickey, Donald E. "A Human-Engineered Reactor Control Panel". Nucleonics, 18:12 (December 1960).

13. Reactor Hazards Evaluation. Oak Ridge School of Reactor Technology Notes. U. S. Atomic Energy Commission

14. Pierce, R. M. "Emergency and Shutdown Cooling Systems of U. S. Power Reactors". Nuclear Safety, 1:4 (June 1960) U.S. Atomic Energy Commission.

15. Research Reactors. U.S. Atomic Energy Commission. McGraw-Hill Book Company, Inc. 1955.

16. Hurwitz, H. Jr., "Safeguard Considerations for Nuclear Power Plants". Nucleonics, 12:3 (March 1954)

17. Greenwood, J. W. "Contamination of the NRU Reactor in May 1958". U.S. Atomic Energy Commission, AECL-850, May 1959.

18. Rupp, A. F. "NRU Reactor Incident". Nuclear Safety, 1:3 (March 1960).

19. McDonald, W. B. and Devan, J. H. "Sodium Reactor Experiment Incident". Nuclear Safety, 1:3 (March 1960)

20. Editors. "Fuel Damage in WTR Incident". Nucleonics, 18:9 (September 1960).

21. Report on WTR Fuel Element Failure 3 April 1960. Westinghouse Electric Corporation. (7 July 1960). WTR-49.

22. Lewis, W. B. The Accident to the NRX Reactor on 12 December 1952. Atomic Energy of Canada Limited, Chalk River Project. (13 July 1953). AECL-232.

23. Hurst, D. G. The Accident of the NRX Reactor Part II. Atomic Energy of Canada Limited, Chalk River Project (23 October 1953). AECL-233.

24. "Reactor Accident at Windscale". Nucleonics, 15:11 (November 1957)

25. McCullough, C. Rogers. "The Windscale Incident". From the Proceedings of the 1958 AEC and Contractor Safety and Fire Protection Conference. TID-7569.

26. Epler, E. P. "HTRE-3 Excursion". Nuclear Safety, 1:2 (December 1959)

27. Editors. "AEC Inspects Reactors for Safety". Nucleonics, 13:3 (March 1955).

28. Larin, Frank. "Reactor Safety System Design Criteria". Nuclear Safety, 1:1 (September 1959).

29. Beck, C. K. Standards for Permits and Licenses (draft). (U.S. Atomic Energy Commission).

30. Harrer, J. M. "Control-Rod Mechanisms". Nucleonics, 13:6 (June 1955).

31. Epler, E. P. "A Program to Evaluate the Performance of a Safety System", Reactor Safety System Response. U.S. Atomic Energy Commission. ORNL-2318

32. Larin, Frank, et al. "Criteria for Evaluation of Reactor Safety Systems Against Their Own Failures". From a paper presented at an American Nuclear Society Special Session for Invited Papers. (6 March 1959).

33. Bohannon, James R. Jr. "Inspection of the Air Force-Convair Nuclear Facility, 28-31 March 1960".

34. Lewis, Valta G. "Inspection of the Air Force-Convair Nuclear Facility, 28-31 March 1960".

35. Mann, Marvin M. "Proposal to ANS Standards Committee - Minimum Standards for Operation of Research and Test Reactors - Organization and Procedure". (6 March 1960).

36. Title 10, CFR, Chapter 1, Atomic Energy Commission. Part 55, "Operators Licenses". Federal Register, 21:1 (4 January 1955).

37. Bohannon, James R. Jr. Notes prepared for use in inspection of Air Force-Convair Nuclear Facility, 28-31 March 1960.

38. Novak, J. R. Radiation Safety Guide. Argonne National Laboratory (June 1956). ANL-5574.

39. Flora, J. W. and Shortall, J. W. "Testing Operating Safety of a Water-Boiler Research Reactor". Nucleonics, 14:6 (June 1956).

40. McCullough, C. Rogers. "Safety Problems". Nucleonics, 12:4 (April 1954).

41. Powell, Robert W. "Problems in Operating a Research Reactor". Nucleonics, 11:10 (October 1953).

42. Beresovski, Theodore. "Safe Design of In-Pile Tests". Nucleonics, 18:9 (September 1960).

43. Personal communication with Valta G. Lewis, Operations Officer, NETF, Wright-Patterson Air Force Base, Ohio on 24 January 1961.

44. Bright, G. O. and Forbes, S. G. Miscellaneous Tests with the SPERT I Reactor. Phillips Petroleum Company, Atomic Energy Division. IDO-16551 (TID-4500). 23 October 1959.

45. Forbes, S. G. et al. Instability in the SPERT I Reactor Preliminary Report. Phillips Petroleum Company, Atomic Energy Division. IDO-16309 (TID-4500). 10 October 1959.

46. Morris, Peter A. "Reactor Inspection". Nuclear Safety, 2:1 (September 1960).

47. Personal communications with Lt. Colonel Claude A. DeLorenzo of the Directorate of Nuclear Safety Research, Kirtland Air Force Base, NM on 16 February 1961.

Supplemental Bibliography

Many reference sources exist in the field of nuclear reactor facility safety and in safety inspection, which were not pertinent to the preparation of this safety inspection guide. However, the best of these additional sources are listed below for the further edification of the reader.

1. Balligand, P. "Reactor Incidents at Saclay". Nucleonics, 18:3 (March 1960).

2. Bell, C. G. and Culver, H. N. "Accident Analysis". Nuclear Safety, 18:6 (June 1960).

3. Bohannon, James R. Jr. "Mockup Tests Determine How Test-Cell Voids Affect Reactivity". Nucleonics, 14:11 (November 1956).

4. Forbes, S. G. and Nyer, W. E. "Role of Reactor Kinetic Behavior in Reactor Safety". From a paper presented at the Reactor Safety Conference, 31 October 1957. TID-7549.

5. Harris, W. B. "Examining the Value of Reactor-Site Environmental Surveys -- A U. S. View". Nucleonics, 18:8 (August 1960).

6. Langer, B. F. "Lessons Learned from Pressure Vessel Investigations". From a paper presented at The Proceedings of the 1958 AEC and Contractors Safety and Fire Protection Conference. TID-7569.

7. Lewis, W. B. and Siddall, E. "Reactor Safety Standards and Their Attainment". (13 September 1957). AECL-498.

8. Lowry, F. C. and Quigley, J. A. "Integrated Safety Organization in Nuclear Plants". From a paper presented at The Proceedings of the 1958 AEC and Contractor Safety and Fire Protection Conference. TID-7569.

9. Robinson, G. C. "Containment Vessel Leak Detection". Nuclear Safety, 1:1 (September 1959).

10. Smith, Richard J. Reactor Safety, A Selective Bibliography. U.S. Atomic Energy Commission, Government Printing Office; May 1958 (a listing of over 400 references). TID-3525.

11. Truran, W. H. "Trends in Radiation Protection with Nuclear Reactors". Nuclear Engineering and Science Conference, 1958. Preprint 49.

12. Weisman, J. et al. "Coolant Temperature Rise in a Pressurized Water Reactor During a Loss of Flow Incident". Nuclear Engineering and Science Conference, 1958. Preprint 137.

13. Williams, Charles R. "Safety Personnel Requirements in Nuclear Energy Plants". From a paper presented at the Proceedings of the 1958 AEC and Contractor Safety and Fire Protection Conference. TID-7569

Appendix A

Abbreviated Reactor Facility Inspection Guide (Ref 3: A.2-10)

The questions of the following checklist are provided as a guide to the nuclear reactor facility inspection team to aid the team members in determining the safety with which the facility is being operated. The guide is not intended to stifle or restrict the inspector; on the contrary the inspector is encouraged to pursue any discrepancy noted during the course of inspection.

A. Organization

1. How are authority and responsibility exercised?
2. How is authority delegated?
3. What are requirements for the operating crew as to number, training, experience?
4. What are the requirements for reports and records?
5. What is the relationship between the operators and the technical group?
6. How are experimentalists and visitors handled by the operating crew?
7. What are the provisions for systematic review and evaluation of the reactor status and performance?
8. What is the function of the hazards or safety committee?
9. What is the general attitude, experience and morale of the group?

B. Check of conventional inspections

1. What were the results of the conventional inspections and tests (ASME standards, etc.)?

2. Were the conventional inspections performed by responsible and qualified people?

C. Physical layout and environment

1. What will be the effect of blast damage on components, radiation control and reactor control?
2. How does physical layout affect maintenance and routine operation?
3. What would be the effect of inadequate conformance to tolerances on the performance of equipment and systems?
4. How much isolation has the site?
5. What facilities exist for the containment of gaseous fission products?
6. What facilities exist for handling radioactive fuel elements and other irradiated components?
7. What are the special features of loops or other experimental devices?

D. Instrumentation and Controls System

1. What are the composition, size and geometry of control rods?
2. What are the number of rods, rod positions, and worth in reactivity?
3. What is the rod drive mechanism?
4. How can mechanical, electrical, or human failure affect motion of rods?
5. What is the safety action of the rods?
6. Are safety and control rods separate?
7. What is the clearance between rods and other components?

8. What fail-safe or other safety features exist?
9. What other methods exist for control?
10. What is method of control?
11. What is the schedule and method for performance checks of components and systems?
12. What is the position indication?
13. What is the nature, strength, location, and distribution of the neutron source(s)?
14. What process variables are measured?
15. What detectors are used and where are they located?
16. How many detectors are used and what is the basis for selection?
17. Are controls and indications properly located and, in general, "human-engineered"? (Ref 12: 80-82)
18. What continuity of measurement, independence and multiplicity exists?
19. What is the schedule for response check?
20. What provisions are there for detection of malfunctions?
21. Are time constants consistent with phenomena monitored?
22. Is all installed instrumentation necessary? (Ref 13: 39-40)

E. Safety System

1. What is the connection system between instruments and control and/or safety rod actuators?
2. What restrictions exist on rod motion during start-up and normal operation?
3. What process variables can cause automatic safety action or annunciation?

4. Is each variable in the safety system backed up by one or more independent system?
5. Is there more than one safety setting for some variables?
6. Is there more than one safety action (e.g. rod reversal, slow shutdown)?
7. How are trip levels for safety action chosen?
8. What minimum portions of the safety system are required for reactor operation? How is this determined and controlled?
9. What malfunctions of the safety system components cause safety action or give a signal of malfunction?
10. Can the operator ignore or override a signal of malfunction?
11. What are the time constants of the safety systems (relay, rod drop, etc.)?
12. What is the effect of flow, pressure, and temperature on the time constants of the system?
13. Are the time constants of the safety systems consistent with the phenomena monitored?
14. What by-pass features can affect the safety system and how are they controlled administratively?
15. What interlocks are incorporated into the safety and how are they controlled?
16. What supplementary safety action may be employed (e.g. poison, liquid level, etc.), and how is such action initiated?
17. What are the power supplies for the safety, control and instrumentation system, and what auxiliary backup power supplies are available in case of failure of the primary supply?

18. If safety rods are separate from control rods, what are the composition, size, geometry, number, worth, and position in the lattice of the safety rods?
19. Are the safety systems really independent?
20. What parts of the safety system are or are not fail-safe?
21. What is the schedule and procedure for the operating crew to calibrate and check the safety system?

F. Analysis of Reactor Physics

1. What is the maximum possible excess reactivity?
2. What reactivity is held in the rods at shutdown and at normal power?
3. Can the core be rearranged to reduce the worth of the rods?
4. What is the maximum rate of change of reactivity and how can this occur?
5. What is the delayed neutron fraction?
6. What reactivity coefficients exist?
7. What is the maximum allowable power level and how is it determined?
8. What are the contemplated fuel and rod exposures?
9. What is the minimum critical assembly in the existing possible environments(air, water, mixture)?
10. What are the effects of movable experiments?

G. Heat Generation, Transfer, and Distribution

1. What are the normal heat generation rates in the core, blanket, shields, components and structures?
2. How are these heat generation rates determined?

3. How is heat removed?
4. What are the normal operating temperatures and how are they determined?
5. What is the maximum temperature and specific power in the fuel?
6. How do the temperatures affect the mechanical properties of the fuel and cladding?
7. What are the limiting temperatures and power levels?
8. If the coolant is lost, can the fuel melt from overheat before or after the reactor is shut down?
9. Is it likely that the standard flow system can be clogged?
10. Is emergency cooling provided?
11. If there is no emergency cooling system, does the reactor have inherent safety or system configuration to provide adequate protection? (Ref 14: 38-40)

H. Shielding and Radiation Control

1. What are the radiation levels during and after normal operation in process and non-process areas?
2. What are the induced radiation levels for components, structures, and equipment?
3. What are the provisions for temporary shielding?
4. What is the schedule and program for routine surveys?
5. How is control of personnel exposure effected?
6. What is the special work permit procedure?
7. What form of and how much waste is there?
8. What is the procedure of waste disposal?
9. What measures exist to protect against inadvertent release?

10. What equipment is available for, and what are the procedures for normal operation?
11. What emergency equipment and procedures are available and what are the provisions for decontamination?
12. For containment, what are the specifications, methods and schedules for tests?
13. What are the barriers between fission products and the edge of the site?
14. What effect will radiation have on instrument or component performance?
15. What are the qualifications of the Health Physics personnel?
16. How are fuel element failures detected and located?

I. Operating Standards, Procedures, and Practices

1. What systems, technical manuals and standards, process standards, general procedures, checklists, and emergency procedures exist?
2. How are normal and abnormal operation defined?
3. How are procedures, standards, etc, constructed, approved, reviewed, modified and disseminated?
4. Who is responsible for the adequacy and implementation of the various procedures?
5. What is the disaster plan and has it been integrated into other AEC, Air Force and Civil plans?
6. What follow-up procedures are required after abnormal operations?
7. How do operating procedures prevent mistakes in calibration of instruments, setting of trip points, and deviations from procedure?

8. Are the operating personnel familiar with the standards and procedures that apply to them? (Ref 13: 39-40)

9. Are all irradiated materials, including test samples, handled in a responsible manner to include maintenance of accurate and complete records? (Ref 1: 149)

J. Maintenance

1. What is the schedule and method for performance checks of components and systems, and are records kept of the results?

2. What system is provided for surveillance, calibration, and checking of the safety system by the maintenance crew?

3. What is the schedule for response check and maintenance?

4. Is access allowed to all instrument areas, most control areas, and the central control room during operation? (Ref 1: 149)

5. What are the procedures for unloading and reloading the reactor? (Ref 1: 149)

6. Is the safe-work permit mandatory before maintenance can be initiated [illegible] the reactor or any of its components? (Ref 15: 275)

7. Are there separate "warm" and "hot" shops?

8. Is defective equipment stored in a sufficiently shielded location or sent to the "burial ground"? (Ref 15: 275)

9. What are the qualifications and experience of the maintenance personnel?

10. Is preventive maintenance emphasized and are records of work performed maintained currently?

11. Is corrective maintenance performed promptly and are current records maintained?

Appendix B

Analysis of Past Accidents and Incidents

Purpose of Accident Analysis

"Present analysis shows that a serious nuclear accident can occur only as a result of inadequate design, gross equipment failure, and human error. At least two of these factors must be present to cause trouble." (Ref 16: 57)

Even these stringent requirements do not rule out the possibility of nuclear accidents, and in fact, several accidents and radiation incidents have occurred in the United States.

The safety record of the nuclear industry in remarkable, however, in that so few accidents have taken place, and progress has sprung from these accidents. Each accident and incident has been thoroughly investigated, and invaluable lessons have been learned by the industry. In order to see what these lessons are and, perhaps, where more vigorous safety inspections might have prevented misfortune, several accidents are discussed in this appendix.

NRU Reactor Incident. The NRU reactor at Chalk River is an engineering and research reactor with a thermal power output of 200 Mw. The reactor core contains 200 fuel assemblies, moderated and cooled by heavy water, with each assembly consisting of five flat metal bars of natural uranium, clad with 1S aluminum. As there had been cladding failures of 13 of the original rods, the sheathing for replacement rods was applied by extruding the aluminum onto the uranium bar.

In order to insure the safety of the reactor, and yet avoid unnecessary shutdowns, most of the components in the control system are provided in quadruplicate. Then if a signal in one channel disagrees with the other three signals, the one channel is disconnected. When two or more channels give signals, the reactor shuts down. In addition, the reactor will shut down due to excess power, excessive coolant temperature, unusual flow conditions, or excessive radioactivity in the coolant.

In the six months of operation from the time the NRU went critical until 23 May 1958, the date of the incident, several problems were encountered and some were still under investigation when the incident occurred. These problems were a high background of activity in the heavy water, power variations due to coolant circulation, unsatisfactory operation of the flow system monitoring in fuel rods, unsatisfactory rod sheathing, and unsatisfactory operation of the system for automatic correction of reactor power readings in terms of exit cooling temperatures. The final system was disconnected and was being adjusted at the time of the incident, but power readings were being taken directly from the ion chambers.

"On 23 May 1958, after a week of steady operation, the reactor suffered an automatic shutdown as a result of excessive power rate-of-rise, but the staff could find no evidence to account for the occurrence. They started the reactor again, only to be met with another automatic shutdown immediately. This time the excessive power rate-of-rise was accompanied by alarm signals indicating a number of unusual conditions in the reactor, the most significant of which was very high radioactivity in the coolant circuit. Some of the other signals were later shown to have resulted from a pressure transient inside the reactor vessel arising from violent failure of one of the fuel rods." (Ref 17: 1-6)

It was postulated that a pressure transient within the vessel occurred due to the bursting of a fuel rod. Then a lowering of the heavy water level could have caused the withdrawal of the control rods. The power overshoot is believed to have been caused by a weak spring on the "low power-normal" selector switch used to initiate the automatic start-up, as some of the necessary electrical contacts would not be made unless the switch were fully turned. Thus, if a minor fuel defect had developed without detection, further attempts to bring the reactor to power would generate a large pressure transient and result in a violent bursting of the fuel rod.

The fact that the original fuel rods had been unsatisfactory would have made it very desirable to have a monitoring system capable of detecting a minor rupture in the fuel rods. This would have prevented the final violent rupturing of the rod. In addition, adequate circuits would have ensured that the proper contact had been made, and this would have prevented the rapid power rise. Such shortcomings in reactor instrumentation are not unlikely, and the nuclear industry can well profit from the frank presentation of their experience by the Chalk River Staff.

In the process of removing the damaged fuel rods, another interesting situation arose. The guide tube could not be lowered completely around the fuel in the reactor. During the process of circumventing this situation, it was discovered that the heavy water had drained out of the flask chamber, thereby leaving no means of cooling the fuel once it was inside the flask. In an attempt by the staff to move the flask to a point where ordinary water could be supplied, the drive motor stopped. This was due to the snout dropping away from the retracted position and causing the motor to stop due to an electrical interlock. Although noticing that the main valve had opened, the crew was unable to close the valve because an electrical interlock prevented this valve from operating unless the snout was fully down. Thus in this situation certain safety interlocks actually increased the hazard by preventing abnormal actions which needed urgently to be performed under the emergency conditions that existed. This points out that special attention must be paid to all safety interlocks to make sure that the created restrictions are necessary and desirable under all conditions. (Ref 18: 70-73)

SRE Incident at Santa Susana, California. (Ref 19: 73-75)

The Sodium Reactor Experiment (SRE) at Santa Susana, California is a developmental reactor intended to improve the sodium reactor concept for civilian power application. The reactor is graphite moderated, sodium cooled, and has 43 fuel element channels with the fuel elements centered in moderator cans. The sodium design outlet temperature is 960° F. The graphite is canned with zirconium to prevent sodium from going into void spaces in the graphite. NaK is used in the annulus to obtain a thermal bond between the uranium slugs and the stainless-steel jacket. Helium is contained in the space above the NaK to allow NaK expansion and to contain fission gases escaping from the uranium.

Several important events occurred during the reactor operating period beginning on 29 November 1959. During run 8 a large range of fuel-channel exit temperatures were observed, and tetralin, an auxiliary coolant, was observed to enter the coolant system. The temperature spread was believed to be due to oxide plugging in the process tubes; the fuel was removed, washed, and returned to the reactor. Runs 9 to 11 showed continuing temperature spreads in the fuel-channel exit, and fission-product contamination showed up in the sodium. In run 13, a high temperature run, after an initial scram and restart, several unusual situations arose. The reactor inlet temperature started a slow rise, the log mean temperature difference across the intermediate heat exchanger started to rise, a thermocouple in a fuel slug showed an increase from 860° to 945° F., some fuel-channel exit temperatures showed an increase, and the temperature difference across the moderator showed an abrupt increase of 30° F. It was decided that a tetralin leak was causing the trouble, and a leak was found in a thermocouple well.

The fuel was examined and found to be dirty, so it was decided to wash the fuel elements. On 4 June 1959 while a fuel element was being washed, a pressure excursion occurred and the shield plug was lifted out of the wash cell. As a result, the fuel washing was discontinued, and the tetralin-cooled seal was replaced by a NaK-cooled unit. In addition, tetralin was removed from the primary system. Run 14 was initiated and some of the same abnormalities of run 13 showed up initially. The reactor was operated at low power until a scram occurred due to loss of the auxiliary primary sodium flow. After re-establishing operation a sharp increase in activity was observed in the reactor room and the stack, so the reactor was gradually shut down. This situation was corrected by replacing the sodium-level coil thimble by a shield plug. Then on 13 July 1959, a manual scram occurred due to a series of negative and positive reactivity excursions, the causes of which were not known.

The preliminary findings of the Atomics International committee indicate the following:

1. Fuel element failures resulted from leakage of tetralin into the primary sodium circuit, from either coolant passage blockage or fouling of fuel elements by tetralin decomposition.
2. The reactor excursion during run 14 was due to the rapid addition of reactivity and the setback circuit failed because it could only handle a slow rate of decrease in the period.
3. As the fuel washing was not completed, the fuel in the reactor

probably contained substantial amounts of tetralin which may have caused the final difficulty.

"It is the opinion of the reviewers that the reactor instrumentation under the immediate surveillance of the operator was inadequate to indicate excessive fuel-element temperatures, the blocking of coolant passages, and fission-product leakage. As a result the operators did not consider such indications (where they existed) serious enough to warrant shutting down the reactor. Since the SRE is a 'developmental facility built to investigate fuel materials,' it would appear that additional instrumentation, as well as closer technical management, might have reduced the damage to the SRE core". (Ref 19: 75) It might be added that such shortcomings might easily have been identified by meticulously thorough inspections.

The Westinghouse Test Reactor Incident (Ref 20: 104-105)

"In the course of a calibration run on 3 April 1960, one of the uranium-aluminum tubular fuel elements in the Westinghouse Test Reactor failed and spread fission products through the reactor cooling system". (Ref 21) Disassembly following the fuel element failure showed that two inner tubes had melted to within approximately 11 inches from its top, and a large amount of molten fuel alloy had flowed out from between the cladding.

Heat transfer tests showed that failure of the fuel element could have occurred only with bond defects greater than one-half inch in diameter. Reinspection of spare fuel elements by ultrasonic means lent credence to this analysis, as defects ranged from 0.015 inch to greater than one inch in diameter. Initial inspection tests, using the blister method, however, had not shown these defects. Westinghouse reported that the reinspected fuel batch showed a total of 1464 defects, with more than 133 defects being greater than one-half inch in diameter, and this was from a sampling of only 237 tubes.

Westinghouse concluded that fuel specifications had been too loose and inspection of fuel elements had been too lax. In order to maintain the increase in hot-channel factors, due to defects, less than 10 percent in future fuel elements, WTR specifications now call for ultrasonic inspection. The standard of inspection is acceptance of only those fuel elements with defects no greater than 0.125 inch. In addition to inspection methods, doubts were raised about the fabricating methods and chemical analyses of samples. It is thought that vacuum melting might be better in eliminating gas inclusions and a radiation-count scan better in determining uranium concentration and uniformity. The feasibility of bending a thick fuel plate 360 degrees was questioned also, as such bending introduces an increased risk of shearing along the clad-meat interface.

Westinghouse has made a very thorough analysis of their problems, as is evident, but the important lesson is the application of their experience. An inspector should look closely into fuel fabrication, inspection, and forming procedures, particularly in a test reactor using new type fuel elements or subjecting fuel to new conditions. In particular, the ultrasonic testing of fuel elements appears to be very desirable.

<u>The NRX Incident</u> (Ref 22: 1-7)(Ref 23: 1-7)

On 12 December 1952, a power surge occurred in the NRX reactor operated by Atomic Energy of Canada, Limited at Chalk River, Canada. The power surge took place during preparations for low power experiments in the NRX. It was determined that the accident was due to a complex concurrence of mechanical defects and human operating errors.

It was fortunate that there were no radiation injuries, for the sequence of events was such that the accident could easily have been more serious than it was. The cause of the explosion was uncertain, but it appeared that the uranium release from the sheath was accompanied by the evolution of hydrogen escaping into the calandria. Meanwhile helium was escaping through holes from the calandria, and a fire was probably ignited by the gas mixture hitting the air. Helium continued to escape until the gas holder dome reached the upper limit of its travel, and then when the air entered the calandria, a hydrogen-oxygen explosion could have occurred.

The basic mechanical defect was due to the complexity of the rod-dropping mechanism and the rod position indicator lights showing the rods had dropped all the way, when they had not done so. In addition, a member of the project had opened by mistake three or four bypass valves on the shut-off-rod air system, thereby causing three or more shut-off-rods to rise when the reactor was shutdown.

This accident indicates four specific lessons from which others may profit:

1. The rod-dropping mechanism should not be needlessly complex with required combinations of action.
2. The safeguard bank of shut-off-rods should be withdrawn soon after shutting down, instead of as the first step of actual start-up.
3. Any reduction of cooling system capability should be based on theory, not inadequate working experience.
4. Inspection should ascertain that current operating practices are governed by design considerations.

The Windscale Accident (Ref 24: 130, 204-5)

The Windscale reactors, I and II, were designed to produce plutonium for the British nuclear weapons program. Windscale I is a reactor in the classic form, a graphite cube with horizontal fuel channels cooled by air exhausted to the atmosphere. The reactor core consists of several hundred tons of uranium in aluminum-canned slugs. Prior to the accident on Monday, 7 October 1957, Windscale had been shut down for maintenance more than 30 times without incident.

During a routine maintenance shutdown on reactor I, the operating crew discovered that several fuel elements were glowing-red hot. An investigation of this abnormal situation showed that the aluminum cladding had failed and that oxidation of uranium metal was releasing radioactive combustion and fission products through the four-hundred-feet-high stack to the surrounding countryside. Urgently needing to extinguish the uranium fire, the Windscale personnel considered using forced air, but discarded this idea due to very grave possibility of causing an explosion. Late that night, Prime Minister McMillian was notified of the impending national disaster. Early the following morning it was decided to try flooding the core with water, and this method proved to be successful, ending the crisis. Although there were no serious radiation injuries, both of the Windscale reactors were inoperative and the financial loss to the government was increased by the decision to purchase and dump all milk produced in the surrounding area for the following several days. This last step was taken as a precautionary measure, although the radioactivity count in the milk was slightly higher than normal.

Generally, the accident was caused by two factors: an annealing operation was underway and the reactor was also being used as an in-pile experimental facility. In order to effect the annealing of the graphite, the reactor core temperature had been raised 200° F. above normal to induce the Wigner effect. Conduction of the in-pile experiment probably further aggravated the temperature problem, and the uranium fire simply got out of control before it was detected.

C. Rogers McCullough, noted authority in the nuclear field summarized the important lessons to be learned from the Windscale accident as follows:

1. Use of the Wigner release annealing method was risky in a reactor not originally designed for this operation.
2. The operation of the piles had been so successful that confidence had built up to a dangerous degree.
3. No significant studies had been made of accidents that could occur during operation of the Windscale pile, and there was no provision of facilities to cope with burst slugs during the Wigner release operation.
4. The means of detecting burst slugs during the Wigner release were inadequate.
5. The methods of measuring slugs and graphite temperatures throughout the pile were inadequate as there were too few thermocouples.
6. There were no means of detecting a graphite fire.
7. For the Wigner release operation there was no written procedure with criteria for steps to be taken in event of abnormalities.
8. Insufficient technical manpower was available to advise the operating crew on abnormal problems.
9. The organization and procedures for dealing with an accident after it occurred were inadequate.

In addition, Dr. McCullough stated that the best way of extinguishing a graphite fire is by smothering it, and that the water method, as used by the British, could form hydrogen and carbon monoxide to cause a violent explosion. (Ref 25: 74-85)

These nine points above clearly illustrate the importance of operational and maintenance procedures, proper instrumentation, adherence to design considerations and proper inspection to insure compliance with safe requirements.

<u>The HTRE-3 Excursion</u> (Ref 26: 57-59)

The Heat Transfer Reactor Experiment number 3 (HTRE-3) was a direct-cycle reactor operated to undertake extensive tests, including an evaluation of the controls, to obtain information for safe operation of the system. The HTRE-3, operated by General Electric, attained criticality on 24 October 1958, at the National Reactor Test Station (NRTS) in Idaho. The test series was abruptly terminated approximately four weeks later by a power excursion which melted several fuel elements and released a large quantity of fission products.

In order to fully understand the causes and significant features of the accident, the chronological sequence of events preceding the excursion are described. The reactor was operating with reduced coolant flow so that heating rates could be measured. Several chambers, including one of three safety chambers, had been replaced with heat-rate sensors, but signals from either of the two remaining safety chambers would initiate a scram. At the conclusion of a 60 kw run on 18 November, with reactor behavior being as anticipated, preparations were made to repeat the test run at a higher power of 120 kw. Since all ionization chambers were inserted to their full-in positions, circuit constants were altered in the two linear-flux channels so the servo could operate over the range of 15 to 150 kw. The scram level was then set at 180 kw.

The final run proceeded normally up to the power range and then control was switched to the servo to increase the power level on a 20-second period. The reactor power increased as was expected, but on a 10-second period, until the desired 80 per cent of full scale on the linear-flux recorder was reached. Then the linear-flux recorder showed the flux level to be dropping rapidly, and the servo, seeing a negative error signal, continued withdrawal of shim rods. This situation continued for about 20 seconds and then the reactor was shut down by means of a reactivity loss of more than two per cent and/or a temperature scram. The reactivity loss was due to the fuel melting and the temperature scram was initiated by melting of the thermocouple lead wires.

The primary cause of the power excursion was attributed to insufficient voltage applied to the chamber terminals due to operator error. It is most significant to note that this cause and other allied circuitry faults caused the flux recorder to indicate a negative period, when, in fact, the reactor period was positive. Thus, even though the servo was correct in withdrawing shim rods, the signal to the servo was totally erroneous.

The following conclusions were reported by the Lockland Aircraft Reactors Operations Office: (Ref 26)

"It is concluded that the primary cause of the incident was the inability of the linear-flux circuitry to indicate true reactor power. This malfunction is attributed to coincident human factors and not basic design of the reactor or the instrumentation. The failure to remove the filters from the chamber power source and the failure to set the specified voltage (1500 volts) at the chamber power source constitute the human elements. The elimination of either of the above could probably have prevented the incident".

"In retrospect, it is conceivable that the incident could have been averted or delayed and damage reduced if the ion chambers had been withdrawn to their design positions of 11 and 14 inches and suitable amplifiers utilized to provide the necessary signal strength".

"It is further concluded that the damage to the reactor may have been reduced had the operator chosen to monitor the hottest thermocouples and the scram trip set at a temperature much nearer the expected maximum fuel-element temperature... for this portion of the planned experiment".

"A possible contributory cause was the operator's decision to go to previously unattained powers on automatic servo control. It is believed that the operator might have recognized the malfunction had the reactor power been increased in small steps under manual control".

The Nuclear Safety reviewer, E. P. Epler, had several comments on the above conclusions and certain remedial actions taken by the General Electric people. The actions of the facility personnel were uniquely directed toward the prevention of only this particular accident, and these actions should be broadened to include other types of failures. The conclusion that the operator would have recognized the malfunction is weakened by the fact that abnormally short periods were introduced on two occasions and no remedial action was taken by the operator.

The imprudence of going to previously unattained powers on automatic control is obvious, but such action is unwise even on manual control, in view of the fact that both the servo and safety systems were operating with chambers working at previously unattained currents. The operator should have used the manual and automatic controls to first test the safety system under the planned test conditions.

Using only identical types of instrument channels at the new higher power levels indicates that identical behavior should occur in all channels.

So, even a planned test would be hazardous because of the absence of other instrumentation on the range of the designed trip level. This suggests that the basic design might have been at fault also. In addition, the practice of employing the same chambers for both safety and control resulted in the safety system failing at the same time as the servo which precipitated the incident. Finally, Mr. Epler concluded that monitors to ensure the continued operability of the safety system are highly desirable.

A comparison of the views presented above, those of the personnel operating the facility on one hand, and those of an academic reviewer on the other, illustrate a very significant point. An inspector who is not closely associated with the facility to be inspected, is often capable of a more penetrating safety analysis than is the man who works daily in the facility. Herein lies one of the most profitable advantages in the safety inspection system.

Vita

Leon E[redacted] McKinney was born on [redacted], [redacted]. His parents are James David McKinney, Sr. and Mary [redacted] McKinney. After graduating from [redacted] in 1948, he attended [redacted] for three years, prior to entering the U. S. Military Academy in 1951. Upon graduation from the Military Academy in June 1955, he was awarded the degree of Bachelor of Science and was commissioned a Lieutenant in the Corps of Engineers, Regular Army. Prior to attending the Institute of Technology, he served in the United States and Germany as a platoon leader, company executive officer, and staff officer in the 23d Engineer Battalion (AD). He is married to Linda [redacted] McKinney and has one son, Leon E. McKinney, Jr.

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This thesis was typed by Leroy F. Anderson

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