INCIPIENT FAILURE DIAGNOSIS FOR ASSURING SAFETY AND AVAILABILITY OF NUCLEAR POWER PLANTS

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Preface

At the request of the AEC Division of Reactor Development and Technology, the Oak Ridge National Laboratory sponsored this unclassified national conference on Incipient Failure Diagnosis for Assuring Safety and Availability of Nuclear Power Plants. The purpose of this meeting was to make available to the AEC and the nuclear power industry information regarding latest developments in the fields of incipient failure detection, on-line detection and diagnosis of component and system malfunction, and measurement of component and system degradation.

We identified and selected five topics related to on-line component testing, nondestructive testing, and in-place inspection techniques as being the areas of concern for assuring the safety and availability of nuclear power plants. These topics are briefly described as follows:

1. On-line testing of instrument systems for reactor operation and protection.
2. Nuclear methods of malfunction diagnosis, including neutron-noise analysis, boiling detection, and reactivity anomaly detection.
3. Methods for detection of incipient mechanical failures of the primary system components and vessel internals.
4. Methodology for building into nuclear power plants and facilities provisions for on-line testing and inspection of the system and components.
5. Case histories of reactor accidents and incidents: (a) methods and techniques that first disclosed the progress of the accident and (b) incipient failure and malfunction detection methods that, if used, would have given warning in sufficient time to prevent these accidents.

Last, but not less important, a large part of the program was devoted to incipient failure detection and malfunction diagnosis methods applied in fields not related to nuclear reactors.

We are particularly indebted to F. Schroeder, Jr., for his keynote address in which the purpose and the objectives of the conference were stated in their proper dimensions. We also express our appreciation to our friends from outside the reactor field for their most valuable contributions.

The cooperation of the AEC Division of Technical Information Extension (DTIE) in publishing the proceedings of the conference is gratefully acknowledged; we cite in particular Miss Ruth N. Bean, Editorial Branch, for her invaluable assistance and skillful work in preparing this volume for publication. We thank G. S. Sadowski, ORNL, who acted as editorial coordinator between ORNL and DTIE.

Our sincere thanks also to Mrs. Jo Anne Plemons, secretary of the ORNL Reactor Controls Department, for her assistance with many details before and during the conference. The carefully planned and efficiently executed work of C. E. Normand (ORNL) on conference accommodations and of J. E. Inman (ORNL) on audio-visual aids is also acknowledged.

Since we considered it important that the proceedings be published as quickly as possible, the authors were responsible for editing their individual papers, and we edited the discussion without obtaining review and approval of the participants. As editors we tried to preserve the informal atmosphere of the conference, but in some instances we made editorial changes and deleted comments of minor importance to reduce the extent of the discussion. We extend our apologies to those attendees who will find their comments deleted, shortened, or unintentionally altered in meaning as a result of our editorial changes.

We hope that this volume will not only be a useful reference but will also serve as a starting point for more advanced work in this field.

E. P. Epler, Conference Chairman
D. P. Roux, Conference Program Chairman
Oak Ridge National Laboratory

November 30, 1967

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Welcome

F. C. LEGLER
U. S. Atomic Energy Commission

On behalf of the Atomic Energy Commission, I wish to welcome each one of you to this conference on Incipient Failure Diagnosis for Assuring Safety and Availability of Nuclear Power Plants and to thank you for accepting our invitation. We extend a special welcome to those who come from technical fields outside the nuclear reactor industry, and we hope your attendance here will be rewarding to you to the same extent that we look forward to your presence here being rewarding to us in the nuclear power field. We are very pleased to have our Canadian colleagues with us.

At this time I wish to thank the Oak Ridge National Laboratory for serving as the host and making all arrangements for this conference. I feel sure the Laboratory will make your stay here an enjoyable one.

This was, indeed, not an easy conference to organize and implement since it encompasses so many different disciplines and so many different parts of a reactor plant system. I especially want to thank Dr. Dominique Roux of the Oak Ridge National Laboratory Instrumentation and Controls Division for his part in developing the conference program and arranging for the speakers.

A few words now about the origin of the conference. The need for this conference arises from the AEC's growing awareness that new methods are rapidly becoming available to plant designers and operators for ensuring ever higher levels of integrity and performance in a wide range of complex operating systems. New concepts, techniques, and equipment for this purpose are being developed in the industrial, military, and space fields and some even in the nuclear power field. The broad application of this emerging body of technical art and knowledge, together with the observation that incipient failure detection and diagnosis are fast approaching the enviable stage of automation, mathematical formulation, and standard terminology, indicates to us that plant and system diagnosis as a distinct area of technology is definitely here to stay and is daily growing ever more sophisticated.

This conference is being held so those of you here with current and future needs for ensuring the highest order of plant safety, integrity, and performance can have an opportunity to learn, at first hand, about new techniques and equipment that are becoming available to meet some of your needs. We believe that the aims of this conference can be met only through the free and open exchange of ideas and past experiences among the participants, and we solicit your individual help to make the conference a successful undertaking.

Theme of the Conference

F. SCHROEDER

Phillips Petroleum Company

As the field of reactor technology has matured to include higher power and higher power density, more complex, and increasingly expensive systems, considerations of both economics and safety have led to increasing demands for high-quality construction and inspection techniques to help prevent unnecessary failures of equipment or design inadequacies. This development parallels, or follows that in other industries. It is commonplace to note that the failure of inexpensive relays or small valves can jeopardize the capital investment in a multi-million dollar space vehicle. We in the reactor field also need improved capability for the detection of minor failures while they are still in the incipient stage to permit, in effect, on-line repair of complex systems. Increased utilization of incipient failure diagnostics is necessary, and inevitable, if our reactor industry is to continue to expand.

In planning this conference on the application of incipient failure diagnostics to nuclear power plants, the sponsors have wisely chosen to make it an interdisciplinary meeting in order to bring a variety of backgrounds, experiences, and motivations to bear on the subject. Some of you are actively working on the development and application of failure diagnosis to reactor systems. Others of you are working in this field in other industries. Still others of us are interested listeners, because we have direct interest in the potential for increased application of these techniques to reactor systems; e.g., my own interest in the subject arises from association with programs of reactor safety research and development. I am sure that this conference will provide many of us with a much-needed awareness of the current status of incipient failure diagnostic techniques.

Applications of incipient failure diagnostic techniques in the broadest sense occur in many fields, in addition to the obvious example of the aerospace field, and in addition to the reactor applications that we are here to discuss. Some of these are far removed from industrial technology. At the risk of being carried away by analogy, I note that in the field of medicine we see substantial effort being given to the improvement of diagnostic techniques by the use of sophisticated detectors which can be attached to, or implanted in, the body, and by the use of computer memory banks to aid in

interpretation of the data. The ability of man to interact with, and respond to, his environment shows promise of substantial alteration by advances in weather prediction based on what, in a stretch of the analogy, is incipient failure detection by means of satellite weather stations, and failure diagnosis using computers in weather forecasting. Even in the field of economics, many experts now claim that the necessary incipient failure indicators are available, and that if proper data analysis is performed corrective action can be taken to minimize inflationary or deflationary swings. One might even stretch the analogy further and note the substantially increasing use of public opinion polls in the arena of politics, which constitutes, in effect, a means for detection of incipient failures of performance in time to take corrective action. One no longer has to wait for election day to receive warning that the use of terms like "brainwashing" may have been unfortunate in certain contexts.

The "far-out" examples I have just used do illustrate one point worth making however with regard to the application of incipient failure diagnostics. Proper and meaningful interpretation of the data obtained is frequently a more difficult problem than the development of sensors to detect the failure signals. I'm sure we will find that to be the case in the reactor applications we will be discussing.

Safe and reliable operation is, of course, an economic consideration for the reactor industry. Central station power plants are large and complex systems having high initial capital cost. The loss of revenue which would occur in the event of even relatively minor accidents in large central station nuclear plants can run to many thousands of dollars per day. Economic practicality demands that exacting standards for safety and reliability of operation must be established and met. We are concerned in this conference with both availability and safety; and the two are certainly interrelated. Because of my own background, my remarks will be biased toward safety considerations, but I do not suggest for a moment that economic motivations are not also of great concern.

In recognition that some members of this audience have not been heavily involved in reactor safety considerations, I want to take just a few moments to indicate in very general terms what our fundamental concerns are with reactor safety to prepare them for the attitudes and approaches which will underlie most of the presentations at this conference. I hope the rest of you will bear with my generality.

Our early and strong emphasis on safety in the nuclear reactor industry sometimes has the unfortunate consequence that some of our critics conclude that reactors must undoubtedly be dangerous because we worry and talk so much about their safety. Why do we give so much attention to safety in nuclear power plants?

The fission process, which produces the heat we use in a nuclear power plant, also leaves behind in the fuel elements substantial quantities of radioactive isotopes or fission products. The existence of this fission-product inventory within the fuel core is central to safety considerations. In reactors of significant power capability (and by this we now mean of the order of 1000 MW electric), uncontrolled release of the fission-product inventory contained within the core could constitute a public disaster. Although we are all aware that many industrial plants contain hazardous chemicals which cannot be released without widespread public hazard, the magnitude of the radioactive sources involved for power reactors, the potential contamination problems with long-lived isotopes, and the genetic

implications of radiation exposures make dispersal of fission products a unique public safety problem. Recognition of this led to the now familiar containment concept which, in effect, attempts to put the reactor inside a bottle strong enough so that, no matter what happens, the fission products remain inside. In reality, there are three bottles--this is the triple barrier concept so frequently referred to in reactor safety discussions. The fission products must escape first the fuel element cladding, then the primary coolant envelope, and third, the containment vessel.

The triple barrier concept, however, runs afoul of another feature of reactors as a power source--the afterheat problem--we can't just turn off the switch and let it cool down. Because of the radioactive decay of the fission products, power continues to be generated at a significant fraction of the normal operating level for a significant period of time, no matter what we do. Unless we can cool the core for a relatively long period of time after the system is shut down, the fuel will continue to heat up, the first barrier, the fuel cladding may melt, and migration of molten fuel might even cause a breach of the third or containment barrier. Because of these implications of afterheat in a large power reactor, it is of course inevitable that loss-of-coolant must be considered as an accident of major proportions, and we recognize that if one postulates a pipe rupture as the initiator of a loss-of-coolant accident, this means that the second barrier, the primary coolant envelope, is breached at the outset.

Many minor accidents can be extremely costly in terms of plant repairs and loss of availability, and we are all, of course, quite concerned with these. But from the standpoint of protecting the safety of the public, we tend to be primarily concerned with two classes of major accidents. (Let me note at this point that my remarks will be focussed primarily on the pressurized-water and boiling-water reactors which currently make up the bulk of our commercial power reactor industry. Similar statements, differing in degree but not in kind, can be made for other power reactor types.)

The first of these two classes of major accidents is the loss-of-coolant accident. One postulates that, as a result of inadequate design, or fabrication errors, or deterioration in service, the primary coolant system envelope may be breached. Depressurization of the primary coolant will occur and, in most cases, the ability to cool the core by normal means will be lost. Unless active and effective emergency means are provided to cool the core essentially continuously and for a relatively long period of time (which might be months), the fuel element cladding will melt, releasing the fission products to the coolant and eventually into the containment volume surrounding the reactor. Continued heat production can lead to migration of the molten fuel which could melt through the primary reactor vessel, and, if unchecked, could cause breach of the outer containment volume.

The second class of major accident involves overheating of the fuel with normal cooling capability present. If, as a result of a variety of postulated failures, an inadvertent perturbation occurs in the reactivity state of the reactor, a rapid power increase, or power excursion, will occur. If uncontrolled, such an excursion can quickly outstrip our ability to remove heat from the fuel. If the magnitude and rate of the perturbation are large enough, and depending on the reactor's neutronic and thermal response, the energy deposition rate in the fuel may outstrip the cooling capacity, cause bursting or melting of the fuel cladding, and release of the fission products to the coolant. The sudden dispersal of hot fuel into the coolant resulting from such failure might produce pressure pulses which could conceivably cause failure of the primary system and thus a loss-

of-coolant accident. For some reactors this pressure, by rearrangement of the fuel may cause a worse excursion. For many reactors proper design to make use of the inherent self-limiting properties of reactor cores, with appropriate backup action by sophisticated control systems, can provide assurance that the consequences of credible power excursions will be limited to damage within the reactor vessel. Such damage would, of course, be of economic concern to the operator, but should not present a public hazard. But for the larger, and hence more unlikely reactivity perturbations, concern does persist that the primary coolant system envelope might be breached as a consequence of an excursion.

This overheating type of accident can also occur if for some reason the normal coolant flow passages are blocked in a portion of the core. Local overheating can lead to melting and core damage. Under the proper conditions, dispersal of molten fuel can lead to pressure generation and possibly to propagating failures. In some reactors the coolant boiling produced in an overheated channel can in itself cause a reactivity perturbation and lead to a damaging power excursion.

In gross generality, those then are the principal concerns which underlie all the emphasis we in the reactor business place on safety.

Turning now more directly to the subject of this conference, I would like to identify the five major activities which the industry and the Commission undertake to assure that the safety of the public is not jeopardized by nuclear reactor power plants, and to indicate briefly how each of these activities illustrates the need for improvements in the technology of incipient failure diagnosis.

(1) The first activity is the design of the reactor and associated power plant systems. Systems are designed so that the probability of accident initiation will be quite negligible. Clearly, a sound approach to such design requires careful attention to problems of quality assurance and inspection, to ensure that the system is constructed as the designer intended and will indeed function in compliance with the design. In the complex systems of concern, it is clear that sophisticated techniques of the non-destructive, in-place inspection and testing are essential to adequate quality assurance, and it is clear that the designer must anticipate this need.

(2) The second activity involves the development of operating procedures which will reduce to a negligible level the probability that control system malfunctions, design inadequacies, or operator errors could lead to significant accidents. Reliable techniques for early detection of equipment malfunctions or degradation, or unanticipated system behavior are an essential ingredient of these procedures.

(3) The third activity involves the provision of engineered safeguards systems in the power reactor system. These safeguards are provided so that, in the unlikely event of an accident, the sequence of events which would otherwise occur can be modified, or interrupted, and so that the consequences of an accident will be limited such that no significant public safety hazard ensues. Examples - ECCS and pressure reduction. If these safeguard systems are to perform their intended function, they must operate effectively and reliably. There is an obvious need for techniques for periodic testing and monitoring of safeguard systems which one must realize may never be called upon to operate in their intended fashion in the life of the plant.

(4) The fourth activity consists of several independent assessments which are made to assure that the basic designs and procedures, backed up by the

engineered safeguards, provide the necessary and appropriate safety margins. This is the complex licensing and regulatory review conducted prior to issuance of construction permits and again prior to operation of a reactor plant.

(5) The fifth activity in reactor safety assurance consists of the inspection and testing, provided by both the reactor operator and regulatory agencies during the construction of the plant, and during the life of the plant, to assure that the safety margins intended by the designer and by the regulator are initially achieved and are maintained during the life of the plant.

In discussing the importance of failure diagnostics to these last two activities--the licensing assessment and the continuing audit, or compliance, function--let me call your attention to the proposed set of general design criteria published by the Commission this summer. These general design criteria were intended to assist designers and utilities in preparing applications for nuclear power plant construction permits. They are presently the subject of much discussion. If one looks over this set of 70 proposed criteria with the subject of this meeting in mind, it is interesting to note that 24 of the 70 criteria have to do, in one way or another, with requirements for the inspection or testing of the primary coolant envelope, reactor protection system, the engineered safeguards, etc. I would like to cite three of the criteria as examples which I believe are particularly pertinent to the subject of this conference. Criterion 25 is addressed to the operability of reactor protection systems. It states, "Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred." Criterion 36, addressing itself to the reactor primary coolant system, states "Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leak-tight integrity of the boundary components during their service lifetime." Criterion 38 has to do with the reliability and testability of engineered safety features in the plants; it says, in part, "All engineered safety features shall be designed to provide high functional reliability and ready testability ... the degree of reliance upon and the acceptance of the inherent and engineered safety afforded by the systems ... will be influenced by ... the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant." There is now a continuing dialogue between the Commission's regulatory agencies and the industry concerning the implementation of additional criteria for inspectability in reactor pressure vessels. Such criteria are indicative of the reasons for the high degree of interest many of us have in the discussions we are having here this week. If criteria such as these are to be adequately implemented, substantial advances are required in incipient failure diagnostics.

Before I leave the subject of the general design criteria, I would like to note one additional point pertinent to the subject of failure detection. This is the "so-called" single failure criterion. This criterion, in simple terms, says that sufficient redundancy and independence must be designed into protection systems and safeguard systems, such that no single failure of any component or channel of the system will result in loss of the protection function. The implication of this criterion to incipient failure detection is obvious. If the single failure criterion is to buy us anything, we must have confidence that the first failure will be detected before the second failure occurs.

In summary then, all five activities which the industry and the Commission perform to assure the safety of reactor systems require the

capability for incipient failure detection. The operator, the designer, and the regulator continue to ask for improved assurance that the reactor plant is performing as designed. Assurance is asked that degradations have not occurred in its hardware or circuitry which could compromise the plant safety or its reliable operation. The operator has need for techniques to monitor, test or inspect the components and the subsystems in the plant to detect the onset of anomalous behavior or instability, preferably on-line. It is important to measure the degree of material creep, corrosion, radiation damage, material wear, heat transfer surface fouling, etc., or detect the presence of sticky relay contacts, shorted wires, extraneous material in the cooling system, etc. Tests and inspection techniques are needed which can provide statistically meaningful assurance that all the protective features carefully designed into the system are ready to respond when needed. I hope that the experts assembled here will give us an improved insight into what equipment and techniques, and what possible design modifications, are available today, or will be available in the near future to accomplish the above objectives. We also hope to learn what additional R&D programs are needed to provide improved tools for incipient failure detection and diagnosis.

Having briefly discussed the principal concerns in reactor safety, and having noted the importance of incipient failure diagnosis to the activities of the designer, operator and regulator, I would now like to turn more specifically to the agenda for this meeting and examine how the subject matter of the various sessions fits together.

The first session concerns on-line testing of operation and protection systems. Reactors are highly maneuverable power sources, capable of very rapid changes in power generation rate. To keep this rate in balance with the heat removal rate of the system, we employ highly sophisticated control and protection systems which guard against sudden power increases or sudden decreases in cooling capability whether they occur because of equipment failures, because of unforeseen physical changes in the system, or result from operator error. We demand a high degree of reliability for such systems for two basic reasons: First is the obvious one of public safety - keep the fission products where they belong. The second concerns the operational inconvenience and severe monetary loss which would result if an expensive reactor core were damaged and if the facility were removed from service for a long time. To achieve the reliability we demand, reactor operation and protection systems must be designed to provide capability for on-line testing to detect the first failure and allow corrective action before the system is required to function. Such testing has been a design objective for many years now. In the three papers this morning, we will hear a review of the current state of technology for on-line testing of operation and protection systems; the description of a specific system; and a discussion of the important subject of how often such testing must be performed to achieve the reliability demanded.

This afternoon, we turn our attention to the internals of the reactor. We know from experience that things sometimes happen inside reactors which affect reactivity, and hence the power generation rate, in ways unknown or unanticipated by the control system designer or the operator. Things also happen which influence the ability to remove heat. Some years ago the conference chairman coined the term "felt-hat accident" to generically describe the situation where some object inadvertently left in the reactor system suddenly plugs a coolant channel or channels. Many examples have occurred involving sight glasses, welding cans, or what have you. I'm not aware of any felt hats, yet. [I think it would be interesting some day to determine whether there is any reactor operator who has not at some time

during the operating life of his plant discovered the presence of items in the reactor coolant system which should not have been there. Fortunately, this usually occurs during pre-neutron checkout. The others we all hear about.] We are thus interested in methods of observing anomalies in reactor power and thermal behavior and inferring that something is amiss before fuel burnout occurs. In many of the incidents we have had involving fuel burnout or other core damage, retrospective examination indicates that proper diagnosis of the information that was available to the operator at the time of the incident could have prevented the incident. In this afternoon's session, several speakers will discuss a variety of techniques which have been developed to permit diagnosis of malfunctions from the information available to the operator.

For example, it is of considerable importance to the operator to be aware of when coolant boiling begins in the core, both because of heat transfer considerations and because voids can affect reactivity. This latter is of particular concern in fast reactors with positive sodium void coefficients. Historically, many techniques have been proposed for the detection of incipient boiling in reactor cores by inference from various thermal, neutron, and acoustic signals. The current state of this technology will be discussed in a paper this afternoon.

In principal, the knowledge of the detailed reactor power history as a function of time permits the operator to determine the reactivity inventory of the core at any time. We all recognize, however, that in today's generation of reactors, problems of fuel burnup, burnable poison depletions, xenon production, and space-time variations in neutron flux make such calculations complex and beyond the ability of most reactor control room supervisors to do with a slide rule or on the back of an envelope. Simple questions, such as: "Have I pulled the rods farther than should reasonably be required?" are difficult to answer. The application of on-line computer techniques to keep track of the reactivity balance and alert the operator to anomalies will be discussed this afternoon in two papers.

Sophisticated analysis of statistical fluctuations in the neutron-level signals can provide important information on the state of the core and on unanticipated changes of the state. For example, there are cases where neutronic signals have given advance warning of mechanical vibrations of reactor internals before fatigue failures occurred. We have three papers this afternoon which address themselves to the utilization of noise analysis, or neutron density fluctuation analysis, as tools for reactor malfunction diagnosis.

Tuesday morning, we take a look at the reactor coolant envelope. While a variety of engineered safeguards systems, particularly emergency cooling systems, are provided to mitigate the consequences of ruptures in this system, it is obvious that the best way to deal with the loss-of-coolant accident is to prevent major failures of the primary coolant envelope in the first place. An important tool in such prevention is provided by improved methods for inspection of the system so that significant flaws are discovered before operation. Another is the hope that reliable techniques can be perfected to detect developing cracks and flaws, so that the plant can be shut down long before major system ruptures occur. The Tuesday morning sessions are addressed, first, to various methods under study which offer promise for on-line detection of incipient failures in the primary coolant system, and, secondly, to techniques for initial and periodic non-destructive inspection of reactor pressure vessels.

Tuesday afternoon we expand our sphere of attention to include the rest of the plant. As noted earlier by reference to the design criteria, all

critical components and systems in the reactor plant must be designed so as to permit on-line inspection and testing. The importance of this subject certainly warrants the development of standard methodology for building into nuclear power plants the provisions for on-line testing and inspection. We will hear four papers addressed to this subject. Provisions for such inspection will likely be costly and I would expect that we may have a spirited panel discussion on the pros and cons of this subject at the end of the session.

Wednesday morning we have an opportunity to hear what our colleagues in other industries are doing in the field of incipient failure detection and malfunction diagnosis. Here we hope for substantial cross fertilization of ideas and approaches.

All of us who are concerned with failure diagnosis and accident prevention can profit from a knowledge of actual incidents which have occurred. On Wednesday afternoon we will hear some actual case histories of reactor incidents. I hope the authors will indicate to us whether, and to what extent, detection and diagnosis could have contributed to preventing the incidents or alleviating their consequences.

Finally, Mr. Borkowski will assemble a panel which will attempt to summarize what we have achieved in this conference. I hope that, in addition to the obvious benefits of exchange of information on the state-of-the-art, one achievement will be a better appreciation for what additional R&D work is most urgently needed in this field. Some of us involved in the planning of safety R&D work are concerned with whether or not enough is being done in this field.

I would like to add one last plea with regard to sophisticated failure detection systems. I have referred several times to the need for reliability in plant operation and protection systems, and I have referred to the importance of being able to interpret the failure indications observed. I hope it is obvious that, if incipient failure detection systems are to provide the benefits we are looking for, then they themselves must be very dependable. I'm sure that all of us who have had occasion to spend any time in a reactor control room have, at one time or another, heard an operator say: "There goes that alarm circuit again; call the technician!"; or "Look at the bumps on that recorder trace, why don't they get that instrument working right?" Gentlemen, I submit that failure detection systems that lead to that kind of conditioned response, will not get the job done. Improvements in our ability to monitor the system and detect malfunctions and flaws before they develop into accident initiators represents perhaps our greatest opportunity to increase the safety and economy of nuclear power plants and thereby to help accelerate attainment of the benefits we all hope to gain by the development of a thriving nuclear power industry.

On-Line Testing of Reactor Operation and Protection Systems

E. Siddall, Chairman

Available Schemes for Testing Reactor Instrumentation Circuits and Protection Systems

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This paper is a review of the various principles used in present day reactor instrumentation test circuitry. The need for testing channel circuits is first taken for granted with later the suggestion presented that possibly this need is not obvious. Granting the desirability of testing, however, the circuits presented are examined for frequency of test, thoroughness of test, independence of circuits and test, and whether the testing is manual or automatic. Emphasis in this paper is placed only on thoroughness and independence of test.

An example first illustrates the principle of a closed loop continuous dynamic test system. A modified HFTR movable instrumentation channel is the example used. The closed loop system is the only one that indicates a malfunction if any portion of the electronics, mechanical system or even the neutron moderation exhibits a change. This malfunction can easily be detected by superimposing a sine wave test signal upon the power level demand signal. There are, however, no complete closed loop safety systems in use today, and all safety systems consist of some form of open loop arrangement.

For simplicity the open loop nuclear safety system channels can be divided into three arbitrary areas, the front end containing the process signal and the detector, the middle consisting of the electronic amplification and signal conditioning equipment up through a trip circuit, and the rear end containing the safety logic and driving equipment for reactor shutdown.

Various forms and placements of test inputs and outputs are then presented. For front end testing, the movable neutron absorption shutter scheme is shown as is a high voltage on-off switching arrangement. The Brookhaven system of superimposing an audio signal on the high voltage line to the detector channels is also illustrated. A monitoring circuit for a regulated high voltage power supply that provides an alarm if the voltage is either higher or lower than a preset value is described to make the

point that the monitoring circuit can be about as complicated as the power supply itself.

Middle area testing circuits can be divided into those that test level and/or period. Test currents or test pulses that may be switched into the inputs of the electronic circuits are the most common test signals used today. These signals in general provide a simulated response at two or three points in the operating range and the problems of proper design of the input "operate-test" switch so that it cannot produce undesirable transients or be left in the testing position is mentioned. Illustrative circuits for level testing, period testing and combination level--period test are presented. Each of the circuits has some form of theoretical limitation to its safety performance, yet all of the circuits in common use to date have performed satisfactorily.

Output circuits for checking the open loop safety channel as close to the scrambling rods as possible are described. Included are the Oak Ridge coincidence magnet, and various schemes for indicating partial current changes in an output magnet or clutch circuit.

Safety logic combined with output circuit drivers and devices, and the testing methods for these portions of the channels are then described for three reactors, HTGR, SM-1 and the proposed system for Consumers Power. The HTGR circuits consist of a dual logic and output configuration using relays for the final power output driver. The two out of three logic block is the only item in the system not duplicated and consequently, has been subject to considerable design scrutiny. As described any single component in this logic may fail without impairing the overall system safety.

The SM-1 system features all solid-state output switching and logic using triplicated two out of three logic and identical dual power supplies and output circuits. A unique feature is an automatic testing circuit which feeds in at the trip circuits and sequentially monitors all channel logic to the output clutch coil.

The Consumers Power safety system is a dual system from the channel output through the clutch coil, with one out of two logic. One path however, consists of all solid-state components and the other path of relay components. Two separate different design power supplies for the clutches are indicated. The principal of separation of channels is carried to the extent of mounting one of the dual systems in a separate fire proof enclosure in a different room from the other one.

An interesting British system for a two out of three combination scanning method using timed pulse programming is described. This method enables not only rapid indication of a failure, but also identifies the location of the failure.

The HFIR system of grouped independent functional channels feeding separate legs of a coincidence magnet to enable servicing of an entire bank of channels is mentioned as a superior system and a complete description left to other papers in the session.

Digital methods are presented as to their present status and future promise. Early digital circuits imitate blindly the corresponding analog functions and the early circuits are considerably more complex than their analog equivalents. An early testing scheme for a digital

channel shows coded pulse injection on the detector high voltage line with coincidence inhibit at the trip circuits if the coded pulses work their way satisfactorily through the entire channel.

Projections of future digital circuit arrangements stem around the power of the computer and the added safety and features that it can promise by not copying analog channels. A British proposal is presented wherein the safety function is transferred from the channel itself to two-cross checking computers. Elementary statistical calculations indicate only one failure in 20 years to be anticipated from this approach. More sophisticated systems are projected with reference to the work at NRU where experience shows that component failures do not necessarily impair the system operation. An extension of the NRU work is contemplated whereby a self-repairing adaptive logic system is envisioned as an adjunct to simplified elementary safety channels having no built in test features. Diagrams of this system are described in terms of today's circuitry.

On-Line Testing of Safety Instrumentation for the HFIR*

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To assure the reliability and operability of a reactor protection system it is necessary to either monitor its performance, test its operability, or both. Many early systems were simple enough in concept and hardware that continuous monitoring circuits could be applied which would give alarms when certain critical circuit parameters were exceeded or when failures occurred which would not themselves result in safety action. Monitoring alone is usually inadequate, because it is not always possible to infer from measurements made in a system under normal conditions what its response will be under abnormal conditions. Consequently, nearly all modern protection systems employ testing to determine their operability under conditions representative of the abnormal circumstances under which they were designed to respond. The maximum acceptable interval between tests is determined not by reactor parameters, but rather by the predicted or measured mean time between failures of the protection equipment. With the current state of protection system technology, the maximum acceptable test interval is frequently shorter than the reactor operating cycle for most power reactors and many research reactors. This situation makes in-service testing a necessity.

The ORNL High Flux Isotope Reactor is a high-power research reactor with a refueling cycle of approximately 20 days. It was not possible to guarantee a sufficiently low probability of failure of protection system components to permit testing only during refueling shutdowns, and the system was much too complex to infer operability by monitoring alone. Consequently, a design was initiated for a protection system which would have not only a high degree of initial reliability, but also would have

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a high degree of in-service testability so that this high reliability could be maintained by rapid detection and repair of failures. It has been common practice throughout the industry for many years to use a minimum of two, and as many as three or four, redundant and independent channels of instrumentation in reactor protection systems. Many reactors, to achieve maximum protection, initiate a scram when any one or more of the channels requests it. Others, in hopes of increased operability, require the agreement of two channels, or coincidence, before scram is initiated. Coincidence techniques were adopted for the HFIR to facilitate in-service testing and not because any reduction in the spurious scram rate would be expected. Coincidence was already widely used by other plants with no apparent reduction in the spurious scram rate. In fact, some performance studies^{2,3} showed that a number of reactors employing coincidence protection systems were experiencing higher rates of unscheduled power reductions than were reactors with noncoincidence systems. This suggests that serviceability is dominated by factors other than coincidence. Nevertheless, there was a trend in the power reactor field toward two-out-of-two coincidence for significant portions of their systems in hope of attaining acceptable serviceability.

At the time the HFIR was designed, no useful criteria were available relative to in-service testing of reactor protection systems. Previous experience at ORNL was mostly with systems employing fail-safe techniques to a moderate degree and extensive monitoring intended to annunciate unsafe failures.⁴ The advent of in-service testing brought about the need for some new criteria to guide both its design and use. The following principles were developed and have become unofficial standards for performance at ORNL.⁵ The first and most obvious criterion is that the tests should include as much of the protective system as possible and should be conducted as realistically as possible. There has been and still remains a tendency for designers to apply testing only to the convenient portions of the system where the task is relatively simple and leave the more inaccessible items, such as sensors, completely unchecked. Economics has a powerful influence in this area. The most desirable form of test is a local disturbance of the process itself so that the sensor may respond the way it must for protective action, and so that the resulting signal may be traced through as much of the system or channel as possible. This ideal is not always possible, but alternatives can be contrived which approach it very closely.

Testing of actuators is necessarily incomplete since the plant would be shut down if they were allowed to complete their function. However, once again, interesting alternatives employing transient and recoverable actuator, or latch, action are possible.

The ultimate objective of the tests is to discover any and all possible failures since reliability cannot be assured if unsafe failures remain undetected.

The second criterion is that the test should simulate the actual behavior of the process variable as faithfully as possible. Sudden, oversized test signals might shock the system into responding, but more gradual changes in a process variable might fail to yield the desired response. Such tests can give misleading or false information about the operability of the system, which amounts to undetected failure.

The third criterion is that the test should not interfere with the correct response of the channel to a normal signal while the test

is being performed; that is, the test signal should be superimposed on the normal signal in such a way that it does not overload, interrupt, or disconnect the normal signal flow. If disassembly or disconnection of the system is necessary in order to perform a test, then proper reassembly following the test is very difficult to confirm.

The fourth criterion is that the testing system should not compromise the independence of the channels of the protective system. The testing system should not increase the probability of a single fault to cause all channels to fail, nor should a failure originating in the test equipment affect the operation of more than one channel. This is a very difficult criterion to meet, because there are many forms of channel interdependence and common influence, both obvious and subtle, over which the designer has little control. Many of the more exotic self-testing schemes that have been proposed and used employing pulsing and sequencing techniques flagrantly violate this important principle.

The fifth criterion is that the test should show not only that no component failures have occurred, but also that no loss of independence or redundancy has occurred. When only one channel is tested, one and only one should respond.

The sixth criterion is that the frequency of tests should be related to the predicted or experienced rate of failures discoverable by the tests.

These criteria relate primarily to testing intended to reveal component failures. There are additional considerations which must be given to the design of a system for testability. There are five different kinds of tests that should be applied:

1. The system should be tested at least once under controlled real conditions to make sure that it will provide the required protection.
2. Certain tests must, in effect, be made continuously where the failures in question might be common to all of the protective channels and are therefore not random events. Some examples of this type of failure are major power outages, fire in the control room, flooding, and removal of components for bench test or maintenance. Such events should be detected by monitoring and "fail-safe" arrangement of ultimate system action, and appropriate action should be taken immediately rather than waiting for periodic tests.
3. The actuator-scrum rod response and operability must be tested with some frequency. Since such tests result in reactor shutdown, they are only allowed at times of scheduled shutdown, and the design reliability or mean time between failures must be made commensurate with this interval.
4. The protection system must be tested to discover component failures that are not revealed by other means. This is the most familiar test system (the criteria for which have already been discussed), and is usually the only type required on-line.
5. All protective systems are to some extent dependent upon calibration stability; that is, they are subject to inaccuracy or a

form of failure as a result of component drift. Calibration must be checked at appropriate intervals related to the experienced drift rate of the particular equipment used. Usually the quality of equipment is high enough that calibration is required no more frequently than during scheduled shutdowns for refueling or maintenance. It is sometimes possible to incorporate rough calibration checks into the tests for component failures, although this is a bonus function and not generally required of the on-line testing system.

The HFIR protective system was designed according to these enumerated principles as faithfully as could be managed within the limitations of the budget, time schedule, and courage of convictions. Figure 1 shows a pictorial arrangement of the HFIR protection system. This is one channel of three similar channels, the other two being exactly equivalent in function. As was mentioned before, the system utilizes three channels in two-out-of-three coincidence. This means that signals from at least two of the three channels must agree that safety action is required before a scram can be accomplished. This scheme is defined as general coincidence, as contrasted with local coincidence. The difference is that in a general coincidence scheme, only channel agreement is required to produce safety action. A channel may have a number of inputs from different process variables such as neutron flux, coolant flow, system pressure, and so on, any one of which may cause the channel output to request a scram. For example, a scram might actually be accomplished by the combination of a high flux signal in one channel and a high temperature signal in another channel. Local coincidence, a common alternative arrangement, is defined as requiring the agreement of two signals that are measures of the same process variable to accomplish safety action. It is immediately apparent that the general coincidence arrangement has a somewhat higher probability of false action as a result of two random failures, but this deficiency, in our experience, is more than compensated for by the relative ease with which tests can be performed. The adequate on-line testing of coincidence elements themselves is an exercise in ingenuity.

There are eight fast scram inputs to each channel of the HFIR protection system. Two of these signals are interdependent and are combined into a single response. High power and low coolant flow are equally disconcerting to the core and, although continuously variable flow was not anticipated, a scram was not acceptable upon loss of one or more of the main coolant pumps. In fact, the system does not scram when all three main coolant pumps are de-energized, such as might occur as a result of a major site power outage. The control system, not shown here, automatically reduces the reactor power to a value consistent with the available coolant flow provided by emergency power supplies. In this case the safety parameter is the ratio of power to coolant flow. To achieve fast response, neutron flux made proportional to power is substituted for the actual power signal. A weighted difference of the two signals approximates the ratio. Testing is done both by increasing the flux signal and by decreasing the flow signal. It was hoped that a neutron shutter could be used to actually increase the flux impinging the ionization chamber as shown, but space would not permit. Instead an additional section of the multisection ionization chamber is paralleled with the main section to increase the output current. In this way all functions are checked except for the flux-current proportionality of the chamber. At normal full power and flow, the increased current is sufficient to exceed the trip level of the flux-flow comparator, and

Fig. 1 HFIR on-line testing.

a trip signal is delivered through the OR gate to the magnet control amplifiers which supply current to one coil of each of the four scram-rod release magnets. The current shutoff in the appropriate coils is monitored by ammeters. The magnet itself acts as a two-out-of-three coincidence element, as will be described later. The second part of the test simulates low flow by momentarily bypassing the pressure lines to the differential pressure transmitter in the flow measuring system. This reduces the flow signal and should give a flux-greater-than-flow trip as before. Again, the process variable is not perturbed as would be preferred, but the sensor is caused to respond in a realistic manner.

There are some supplementary tests and monitors of interest in the flux-flow system. Since a neutron shutter or chamber add-on section will produce a satisfactory test only at full power, a prestartup test is provided by superimposing a slow current ramp on the ionization chamber signal. This current yields both a trip test and a rough calibration check of the channel. It is applied through an extra lead into the ion chamber so that a proper response gives some assurance that the ionization chamber is not disconnected. The chamber polarizing voltage is continuously monitored by a similar auxiliary lead from the chamber.

The neutron leakage from the core changes over a core life because of rod shading and changes in axial flux distribution. As a result, the ionization chambers that measure this leakage flux deliver a signal that is not strictly proportional to power. This is true to some degree in all reactors, but the problem is unusually severe in the HFIR because the control rods effectively surround the core. To achieve adequate protection for all conceivable accidents, both the slow but accurate heat-power signal and the fast but less precise neutron-flux signal are needed. A scheme, which we call flux reset, was devised to continuously compare and recalibrate the neutron flux signal to agree with the calculated heat-power signal. The recalibration is done slowly so as not to interfere with response to fast transients. This calibrating system is monitored so that most failures would be immediately apparent to the operator.

A second safety signal is derived from the neutron flux channel by taking the time derivative of flux. In spite of its apparent similarity, this signal is totally unlike a reactor period signal derived from a logarithmic channel. It is simply the rate of change of flux or power, and since it is derived from a linear instrument, it is insensitive below about 5% of full power. Its sole purpose in the HFIR is to provide an anticipatory scram to enhance the protection against the sudden insertion of reactivity at full power, such as might result from a void in the trap region. The trap, or central target, region has a positive void coefficient such that an air bubble which might be carried into the region by the coolant flow would cause a very sudden increase in power.⁶ The flow velocity is about 50 feet/second in the core. The trip level is adjusted to about 20% power per second. In testing, this circuit responds to the chamber add-on test, and in addition, a fast current ramp can be injected to give a trip and a rough calibration.

The next channel protects against loss of emergency coolant flow. When main flow is lost, three pony pumps provide emergency flow of about 10% normal full flow. If the flow drops below the equivalent of that provided by two pony pumps, a scram is requested. The differential pressure cell that provides this signal from the venturi is much more sensitive than the main dp cell and is grossly overranged at full flow.

To test this dp cell at full flow, it was necessary to add an orifice in the line because of the strong pumping action of the venturi. The test, as before, is accomplished by bypassing the dp cell with a solenoid valve. The orifice is sized to give a rough calibration check of the trip point.

The next two tests are probably the most elaborate of the entire system. Functionally they are quite simple, but quite a lot of hardware is involved. The reactor inlet temperature sensors are located downstream of the primary coolant pumps. They are direct-immersion resistance temperature bulbs. The function of the inlet temperature scram is to anticipate the subsequent high outlet temperature which would result as a consequence of failure of the secondary heat removal system. The test of this function is accomplished by actually perturbing the process in the vicinity of the sensor. Hot water is injected into the pipe so as to flow around the individual sensors. The injectors are located downstream so that normal flow over the sensors is not obstructed. Figure 2 is a photograph of a typical sensor-nozzle assembly. Figure 3 shows a test setup used to observe the flow patterns around a sensor using downstream injection at velocities of about 30 feet/second. Figure 4 is typical of the flow patterns obtained. The final nozzle design was a slit as shown in Figure 2.

Fig. 2 Temperature sensor.

Returning to Figure 1, a similar hot-water injection system is used on the reactor outlet temperature sensor. The outlet temperature signal is not used directly for safety action, but rather is used along with inlet temperature and coolant flow to compute heat power. Raising the temperature at the outlet sensor increases the apparent differential temperature across the core and consequently increases the heat power signal to effect the channel test. It may be noted in passing that the heat-power signal is momentarily reduced when an inlet temperature test is performed. This tends to inhibit a heat-power scram for the duration of the test in violation of one of the design criterion. In this case

Fig 3 Nozzle test rig.

Fig. 4 Nozzle flow pattern.

it was acceptable, because the heat-power scram is essentially duplicated by the neutron-flux scram, and the overall ability of the system to respond to high heat power is not inhibited. In fact it is somewhat reassuring to see the heat-power signal respond in both directions to successive tests.

The next safety parameter is vessel pressure. If the pressure should drop too low, boiling may commence and lead to fuel burnout. The detector is a simple Bourdon pressure switch located on a small line leading to the pressurized system. An orifice is placed in the line, and a test is accomplished by opening a solenoid valve which vents to the let-down system. The resulting flow through the orifice produces sufficient pressure drop to cause the pressure switch, which is downstream of the orifice, to actuate.

The last protection system input is called a failed-fuel-element detector. There is some probability that, even under otherwise normal operation conditions, local blockage of flow could cause partial fuel-element burnout or rupture, and it is desirable to scram as soon as the failure can be detected in order to avoid cumulative damage and excessive system contamination. Such an occurrence would be accompanied by a sharp increase in the gamma activity of the effluent water. The gamma activity is normally quite high because of the transient ^{16}N in the system. An ordinary gamma-sensitive ionization chamber is used as a detector and is placed in a convenient location outside the primary shield near the effluent water pipe. The ^{16}N background provides a convenient test source. A lead shutter mechanism is normally in place in front of the ionization chamber, and the amplifier sensitivity is adjusted for a suitable on-scale reading at full power with a clean system. When the lead shutter is opened with a motor-driven mechanism, the flux at the chamber increases sufficiently to produce channel trip.

Each parameter is compared with a reference or another parameter to determine the trip, or actuation, level in a stable, highly reliable comparator circuit. The outputs of these comparators are bilevel logic signals. The abnormal, or trip, level is zero volts and the normal level is minus 10 volts. The OR gate has the characteristic that its output will be zero if any one or more of its inputs is zero; thus channel trip will result if any of the eight inputs reaches the trip level.

The zero output from the OR gate causes each of the four magnet control amplifiers connected to it, one for each rod, to switch off. Each of the magnet amplifiers controls current in only one coil of its associated scram release magnet. Each magnet is itself a two-out-of-three coincidence element. The magnets have saturation characteristics as shown in Figure 5. With only one coil energized nearly all of the developed flux is diverted by the shunt, and practically none couples the armature, resulting in negligible holding force. The shunt is just saturated with the one coil energized. When the second coil is energized with the same number of ampere-turns, nearly all of the additional flux is coupled to the armature since the shunt is now saturated, and the required holding force is developed. Energizing the third coil develops very little additional holding force, because the armature and yoke begin to saturate. In normal operation all three coils are energized. When any one coil is de-energized, as would result from a one-channel trip, the holding force is reduced by only a very small amount. When any two coils are de-energized, the holding force is reduced to near zero and the armature is released and unlatches the safety rods.

Magnet with Saturating Shunt.

Fig. 5 Three coil magnet.

Placing the coincidence elements in the rod mechanisms allows rather complete testing, lacking only a demonstration of the ability of the rod to drop. A successful test is determined by observing the currents decrease to zero in the appropriate coils, one for each rod, and no others.

I have described the routine on-line protection system tests of the HFIR. In addition to the routine tests there are many provisions for nonroutine testing, trouble isolation, and more precise calibration checks. In fact, with the general coincidence arrangement and good channel isolation, the ability to test and repair is limited only by accessibility. One of the principal benefits which has been derived from this type system, which was not wholly anticipated during the design stages, is the relative ease and comfort with which on-line maintenance and modifications can be accomplished. This will certainly contribute to continued high reliability over a long period of time.

There are some additional inputs to each channel which are called slow trips; their combination is called slow scram. These are signals of lesser importance which do not have fast response. The slow trip is accomplished by interrupting the power to all of the magnet control amplifiers in a given channel. Typical slow-trip functions are the operator's manual scram switch, continuity circuits which assure that certain vital components have not been removed from the system, and certain local power failures.

Figure 6 is a photograph of the control room of the HFIR. The protection, regulating, and startup systems, three channels of each, are contained in the aluminum vertical panels directly in front of the console. Approximately the upper third of each is devoted to the

Fig. 6 HFIR control room.

protection system channels. The on-line test pushbuttons are located on the sloping, right wing of the console. There are eight buttons per channel for the eight tests I have described. The additional buttons are for similar tests in other-than-normal modes of operation which I have not discussed. The magnet coil currents are displayed by the panel meters in the top row of instruments on the vertical board. The annunciators above and other panel meters below are also used in determining tests results.

The regulating system, which I will mention only briefly, is also arranged in two-out-of-three coincidence in that any one channel may fail in any manner with a negligible effect on the controlled power. On-line testing of the regulating system is unnecessary, because unlike the protection system, nearly every failure results in an immediately apparent abnormal indication. Such things as paralleled power transistors are checked at appropriately infrequent intervals.

It is quite apparent in examining the system that it was either not possible or not economical to completely fulfill each of the criteria in every detail. The omissions and violations are few and relatively unimportant, however, and the performance is very gratifying.

The reactor has been in power operation now for about a year and a half. In that time no failure to danger has been detected by the testing system so that the on-line test interval is still based on a predicted rather than a measured failure rate. The predicted failure

rate is one per channel per year and the corresponding, very conservative test frequency is once per day. This rate will probably be continued so that sudden changes in failure rate will be discovered quickly. If, for some reason, it should become impossible to test during a cycle and repair is not possible, operation may be allowed to continue for several days without testing, provided the trouble is well understood and does not compromise channel independence.

A high degree of channel independence and the separation and identification of the protection system apart from other functions, are the keys to achieving not only high reliability, but also good testability and the ability to perform on-line maintenance. Even the mathematical assessment of reliability is vitally dependent upon the assumption of independence of channels, and the mathematical arguments are of little value if independence cannot be achieved.

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The Frequency of On-Line Tests for Nuclear Protective Systems

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I wish to discuss the development of a criteria for establishing the frequency of on-line testing of nuclear power plant protective systems. The criteria that I propose will be based upon applying statistical reliability principles to the observed performance of the protective system. But first, it will be necessary to establish a purpose for on-line testing. I suggest that, in essence, the purpose of on-line testing is to assure (make possible) the maintenance of the protective system's required reliability. Reliability may be defined as: the probability of the system performing its intended function when called upon to do so.

I want to call your attention to two key elements embodied in the purpose. First: we recognize that there is some quantitative value for the system's reliability which we deem unacceptable and another value which we judge acceptable and therefore required as the limit for the probability of system failure. Second: we recognize that the system cannot repair itself. Man must interpret test results and take some sort of action to maintain the system. Thus, by providing information and guidance, on-line testing assures or makes possible the maintenance of the system. It should be self evident that if the on-line testing scheme is inadequate or misused, system reliability will degenerate.

Having established a reason for testing, I will turn your attention to the subject of the test frequency. Let us assume that we have three identical systems, built, installed, and operating, and that we wish to determine their mean-time-between-failure (MTBF). Further, these systems are expected to have failure rates of one failure a year, and are to be observed in operation for no more than 10,000 hours. We shall also assume that just at the end of 10,000 hours, one of the systems

fails. Now, what can we say about the MTBF for these three systems? Using small sampling techniques we can determine with 95% confidence that each system's MTBF will be at least 6300 hours. Had there been no failures in this period, the MTBF would have been 9900 hours at the 95% confidence level.

Now, the point to be made here is that the MTBF of a small number of systems can be determined by testing the systems for a limited period of time and by applying small sampling statistics to the observed results. Let's not bother with the mathematics but accept, for the moment, that there are statistical methods for performing such calculations.

I propose that we consider each installed and operating protective system as being part of a continuous reliability test in which the MTBF of the individual systems is continuously calculated. Knowing the observed failure characteristics of the system, the time between tests would then be adjusted to maintain the odds against system failure to a predetermined value. This is possible because the system's reliability is an exponential function of the time between tests (our mission time) and the observed MTBF.

The criteria for the frequency of on-line testing might be as follows: The test frequency and the reliability of the protective system, calculated from the observed MTBF determined at the 95% confidence level, shall be continuously adjusted and maintained so that the chances of the protective system's decaying to the "cannot trip" state shall be less than or equal to 10^{-4} .

Bear in mind as we proceed that it is the plant operator who will have to implement and use this criteria and that he should not have to be an expert in statistical reliability. Therefore, the mathematical model and all the operations involved in applying the criteria should be as simple and straightforward as possible. The ideal model might consist of one or two general equations describing the system's reliability in terms of the probability of success; these in turn could be reduced to a "cook book" of tables.

To implement the criteria in an operating nuclear plant, we would first establish a time zero, the point in time when the system is considered to be operational and no longer in the installation or shake-down phase. Henceforth, the cumulative operating time of the system would be continuously available. An initial test frequency would be established from the designer's estimate of the expected MTBF. The initial test frequency we shall call the designer's test frequency. This would be employed until the observed failure rate of the system permitted or required a change. Upon detection of a protection system failure, the total system on-line time would be noted, the mathematical model or tables and graphs referred to, and the new test frequency calculated.

The complexity of the mathematical model will certainly depend upon the complexity of the physical system. (A simple model will result whenever the system can be divided into valid, independent, and identical protective channels.) Usually one will be able to determine that a channel originates, without ambiguity at a particular point. However, the terminus of a channel may be very difficult to describe in explicit

terms in either the physical or mathematical sense. Hopefully, the system's physical design would permit us to reduce mathematically each system to a set of simple black box probability functions. A clear relationship in a particular sense must exist between the physical system and the mathematical model. For instance, when a failure occurs within the system, the operator must be able to determine without hesitation the mathematical equivalent channel in which the failure occurred. Thus, all elements of the physical system must be identifiable and must lie within the bounds of the model. Conversely, the operator must be able to recognize the bounds of the physical protective system. The designer, and thus the system design, controls whether or not these conditions can be met.

The system designer is vital in making the proposed criteria practical. Should we demand a reliability for the on-line system in excess of the reliability designer into the system, we would be in trouble. We cannot set operating reliability goals without also setting compatible design reliability goals. The relationship of the system design to the implementation of the test criteria is dramatically underscored in the construction of the mathematical model.

The test frequency criteria, as previously stated, must be supported and linked with a system's reliability design criteria. The following is proposed as satisfying both design and test frequency requirements:

The protection system shall be designed to a reliability level so that the probability of the system's failing to perform its intended function when called upon to do so shall be equal to or less than 10^{-5} . The design reliability level shall be computed from two factors: the best available failure rate data for the system's elements at the 50% confidence level and a mission time equivalent to the initial test frequency proposed for the on-line system.

The designer shall develop a mathematical model of the system relating all the system's elements to the system's probability of failure. The end use of the mathematical model shall be to permit the implementation of the following test criteria: The frequency of on-line testing of the protective system and its elements shall be adjusted and maintained so that the probability of the system's decaying to the "cannot trip" state from failures detectable by testing shall be equal to or less than 10^{-4} when derived from the observed system elements MTBF at the 95% confidence level.

The criteria may be summarized as follows:

1. The designer's level is a probability of 10^{-5} based on failure data at the 50% confidence level.
2. The user's level is a probability of 10^{-4} based on failure data at the 95% confidence level.
3. The designer would be permitted to employ any means at his disposal to derive failure data.
4. The user has no choice but must use actual failure data from the installed system.
5. Only those failures commonly called unsafe and detected solely by the on-line tests are to be counted.

The criteria requires only a particular kind of failure to be counted. It is frequently referred to as an unsafe failure because it degrades or blocks the intended protective function of some portion of the system and silent because it does not signal its presence. I call this a silent, unsafe failure. As such failures can be discovered only by on-line testing, it is the primary task of the on-line test scheme to discover their presence.

The designer's reliability goal is more stringent than that specified for the user. The best way to compare the two is to consider a fictitious system which is permitted one unsafe, silent failure in 10,000 hours at the designer's level as against the system with the same observed real failure rate. The criteria would require a time between tests at the designer's level of one week and at the user's level, 2-1/2 weeks, a difference of 2.5.

In actual practice, this system would be placed in service using the designer's test frequency of one week. The designer's test frequency would continue in force until the system's observed failure pattern proved the frequency to be either too conservative or too optimistic as judged by the user's test frequency criteria. If the system performed as predicted by the designer, the user would be permitted to increase the test frequency after approximately 4,000 hours of operation. At the end of 10,000 hours, the time permitted between tests would reach 2-1/2 weeks, provided that no more than one unsafe, silent failure had been observed. Had a silent unsafe failure occurred at the end of the first 3,000 hours of operation, the user's test frequency criteria would have forced a reduction in the time between tests from 7 to 5 days.

The designer's reliability level includes a safety factor if for no other reason than good engineering practice; however, there are other more practical reasons. The designer cannot be expected to account for all factors affecting the system's reliability...therefore, he should design the system to the highest practical level, the ultimate reliability being controlled by unaccountable factors. I believe the probability of 10^{-5} as a design goal to be very near the practical limit.

I also believe that the odds are heavily against two or more identical systems installed in different nuclear plants exhibiting the same failure characteristics over the same period of time. Each installation is unique in a way that cannot be accounted for by the designer.

The designer's reliability goal, in the last analysis, is to provide sufficient safety factors so that the user's reliability goal can be realized for the on-line system. The criteria proposed here gives the design a minimum safety factor of 2.5. Individual designers may elect to employ greater margins.

Once the system is installed and operating, the user's test frequency takes charge. Henceforth, we are no longer interested in the designer's reliability goal but in determining the actual reliability characteristic and maintaining it to an acceptable level. All the unaccountable factors faced by the designer will now manifest themselves in the system's observed characteristics. The failures observed are real and representative of the system. Our task now is to establish the system's reliability to a high degree of confidence.

The criteria requires the user to calculate a MTBF which will be exceeded 95% of the time. From the MTBF (95% LCL) a test frequency is computed so that the probability of failure is equal to or less than 10^{-4} .

I am not prepared to justify the probabilities of 10^{-5} and 10^{-4} used in the criteria statements. The best argument in favor of the 10^{-4} probability at the user's level is found in Vol. 6, No. 4, Section III of Nuclear Safety in an article by E. P. Epler. "The readily obtainable safety-system-failure probability attributable to simultaneous component failure of 10^{-4} , when multiplied by the excursion arrival rate of 10^{-4} per hour, results in an uncontrolled excursion rate of 10^{-8} per hour. This is a factor of 100 better than the 10^{-6} per hour arrival rate of uncontrolled excursions actually experienced and which were attributable to performance failure and not to component failure."

A number of questions have been repeatedly asked with regard to implementing the criteria. I shall now consider several of these.

How well can a protective system be described by a mathematical model? Almost perfectly, depending upon how involved one wants to become with the mathematics. I think that the real problem is judging how far it is practical to go with the model. For example, from a practical standpoint the designer could spend a great deal of time trying to devise a model of a relay that would describe the lack of isolation existing between the relay contacts and the coil. Most of us assume that a relay coil and its contacts are electrically independent and isolated. This is not absolutely true; however, in most relays, it is next to impossible to assign a value to these factors. The designer is wasting his energy in such efforts. Such factors fall into the unaccountable category and may be treated as a part of the uncertainty of his calculation.

May elements of the system be tested at different frequencies? Can the system be modeled to permit the use of different test frequencies for different elements in the system?

The answer to both questions is: Yes. The physical arrangement and operation of most nuclear plants imposes a limitation upon the practicality of testing the whole system at one frequency. The control rods cannot be dropped as part of an on-line test without shutting down. The designer may wish to limit the test frequency on certain black boxes for other reasons, and this should be permitted. I believe that the problems of bookkeeping and complexity of developing a model which will permit this sort of thing will tend to influence the designer to avoid splitting the system as much as possible.

If we are to continue to stress the need for and the effectiveness of on-line testing, then we must have a logical and valid criteria for the test frequency. The criteria I have proposed goes directly to the heart of the problem: "The purpose of on-line testing is to assure the maintenance of the protective system's required reliability."

We should, I believe, begin to use either this criteria or one similar setting the reliability, confidence levels, and probability of failure to values which we can reasonably justify from experience.

Discussion on Papers by Schultz, Anderson, and Patterson

SIDDALL (AECL): I will exercise the Chairman's privilege of opening the discussion. In connection with the first two papers, I would like to mention that on the two power reactors now working in Canada we decided that we should have the shutter test of the ion chambers, because this is the only way to detect saturation of the ion chamber and power supply combination and masking of the neutron signal by gammas at very low power levels. All other methods of injecting electrical signals simply miss this point, and I believe that experience has shown that the electrical parts of the circuits are usually much more reliable than this front end.

If the shutter is placed alongside the ion chamber (as shown in Fig. 1) instead of in front of it, we have a much more convenient geometry. In other words, we merely have to run a test rod out along the same general line as the cable, and thus we can really put the test facility wherever we could put another ion chamber.

Figure 1

Figure 2 shows, as emphasized by Schultz, that there are "pure" and "impure" sections of the system. In our present two reactors, we scram the reactor by dumping the heavy-water moderator. In fact, this dumping is done by a helium flow down through the nest of valves shown in Fig. 2. Each of the three channels goes to a pair of valves, stag-

Figure 2

gered as shown, so that at the moment we have a pure rear end. We can actually test right through the system—all the way to opening the dump valves wide. We do not dump the heavy water if we test it one channel at a time.

Unfortunately, in our future programs we have decided that we cannot really continue to use the dump method. The future reactors will be so large that the economics of this scram method is very poor. We are having to go to a shutoff rod scheme (Fig. 3). The principles that I emphasized before must be followed now. We should have a lattice and

Figure 3

not a chain. In some of Schultz's illustrations, it was shown that the channels came down to one pair of wires before they fanned out into a number of shutoff rods. We believe, in principle, that each shutoff rod should look at all three trip lines and make its own decision whether there is a 2-out-of-3 signal, and so we have a test facility in there. First of all, there are meters to show that the single-channel trip has gotten through all the 2-out-of-3 circuits. As Schultz points out, this does not prove that the rod can get in all the way, but it is a pretty strong help toward showing that everything is all right. The test facility that we are working on is intended to drop the shutoff rod about 4 ft in our Pickering reactors.

The advantage of using a clutch and a winch-and-cable mechanism, rather than a direct-lift magnet, is that we can reenergize the clutch, stop the fall of the rod, and take it back again before there has been any very large reactivity effect. The small reactivity effect

which we do get, of the order of 0.02%, will give a little reaction from the regulating system, which is another assurance that everything is working all right.

LEGLER (AEC): Mr. Anderson, you have a lot of sensors in your on-line testing scheme. Have you noticed any degradation in the calibration or time constant of any of your sensors?

ANDERSON (ORNL): Yes, we have a predicted loss of sensitivity of the ionization chambers because they are neutron sensitive. They are boron lined and under continuous exposure to neutron flux, and so the boron burns up and the chamber sensitivity gradually decreases. The continuous calibration automatically compensates for this. In addition, at appropriate intervals, we can reposition the ionization chamber with respect to the core in order to compensate for the boron burnup.

In the other sensors, as far as I know, we have not detected any significant degradation of the calibration characteristics because none of these sensors is subject to burnup in a similar manner.

LEGLER: How about the time constant of the temperature sensors? Do corrosion deposits have any effect?

ANDERSON: There has been no noticeable response-time degradation. The corrosion is very small, and the temperature-sensor time constants are already quite large. For fast response we use the enhancement techniques where neutron flux is made proportional to heat power to avoid the necessity of having very fast response in the temperature sensors.

DEBOISBLANC (IDAHO NUCLEAR): Mr. Anderson, in connection with the magnet, there is a danger in the idea of the testability of a magnet. The ingenious yoke and shunt system which you described is a very good thing for ensuring on-stream time, but, in connection with the figure of merit for the magnet and release system collectively as a protective device, you must include the latch and other mechanisms or mechanical components (e.g., control rods) which themselves have to do a job. In this connection, your test scheme does not monitor the releasability of the ball mechanisms, for example.

Further, perhaps the most important protective aspect of a magnet and the reason one uses a magnet is the release rate. I maintain that the use of any flux regulation system, such as a shunt, tends to obscure the internal shorted-coil defect, which is the most dominant way of increasing the release time.

Would you care to comment on this?

ANDERSON: With respect to the actuator and release time using the magnetic shunt, admittedly this degrades the ultimate release time of the magnet assembly somewhat. However, if you are interested in numbers, for any combination of 2-out-of-3, or 3-out-of-3 (there is a slight difference), the maximum release time of this mechanism is 12 milliseconds. We measure the release time during reactor shutdown. This was one of the five types of tests I described: the actual ability of the mechanism to release the rod as required is tested during shutdown intervals, and the ability to release is redundant because the 2-out-of-3 now is separate for each rod. Any one can fail without affecting the ability to shut down the reactor.

DEBOISBLANC: I was not really challenging your contention that the shunt itself did not add enough delay to cause concern. What I was saying is that, if one has a magnet system and sets the thresholds for release with the single coil, then, if the current fails or if the turns burn out or if there is an internal appearance of water (leak into the system), the system drops the rod. On the other hand, if there is a shorted turn, then the release-time checks should detect the shorted turn; and, if there are multiple shorted turns, this would reduce the magnetic flux below the threshold level, and the rod would drop. So I fear that this approach tends to decrease the ability to detect the appearance of a shorted turn through its consequent effect on release time.

It might be that, if in your composite shutdown type of test you measure the release time separately from everything else, you are testing the whole system. But that is something else from what you were describing in your system.

ANDERSON: I think I see the difference. Basically, it amounts to the insensitivity of the shutdown test for release time to detect shorted turns in the presence of the shunt.

DEBOISBLANC: This bears on the requirement of the frequency of that particular test because, in the accident analyses, one must not only guarantee that he has a sufficiently rapid release time by design but also that he maintains it during every hour of operation of the system, since one cannot predict when the maximum credible reactivity insertion will occur.

You do, then, rely upon shutdown interval tests and feel that the shunt does not increase the delay time intolerably, that you can check this in the shutdown release time, and that this combination is satisfactory?

ANDERSON: That is correct. I believe your fear is, in this case, unwarranted because we did, in early stages, detect a failure that was not a shorted turn, although it was somewhat like a shorted turn. One of the switching transistors became leaky, and we did detect this by an anomaly in the response time. By testing deeper than is normal for the on-line test, we got to the bottom of this trouble. So there still remains a respectable sensitivity to this sort of thing.

DEBOISBLANC: Mr. Patterson, your statistical approach appears to be largely dedicated to the components, and particularly to the instrumentation and protective system. I wish to question this discrepancy between the figures (the variances, for example, and the probabilities of nonfailure) from the point of view of the designer as opposed to the operator of the system.

Let us assume that a designer sets out to design a plant and finally designs a fuel element. He is responsible for the mechanical design of this fuel element, and he has to take into account all kinds of things relating to the operating performance of this thing: errors in the hydraulics of the system, variances in the manufacturing of the fuel elements, and deformations which are systematically possible in the operation of the reactor throughout its life, and so on. This leads to an interesting problem: the designer must use a conservative design because he must guarantee performance of the system he designs, regardless of the consequences or regardless of the manufacturing technique. If he uses statistical quality control in the process of purchasing many fuel elements, he has some data on what the variances really are. He understands that a standard or a mean deviation is very different from a maximum possible deviation, and this makes a whale of a difference in the ultimate error propagation effect.

SIDDALL: Would this not be a better subject for private discussion with Mr. Patterson? We are short of time now.

DEBOISBLANC: Can I make my question more specific? How does one rationalize this designer-user difference in attitude?

PATTERSON (B & W): That is specifically why the criteria were proposed in the way that they were. I recognize that the designer is faced with unaccountable factors, particularly in component design and equipment design. There are certain things we do not know too much about in the way of reliability, and there is a significant disagreement between equipment designed to a particular performance level and the manner in which it actually performs. The idea here was that with the actual system we can really measure what the system is doing and therefore operate it on that basis. The designer then has a way of checking what he has designed. Once the two attitudes merge, we cannot improve our design ability. If the designer uses reliability and never gets any of the results back out of the field, he does not know whether what he is doing is right or not. He only suspects.

BOYD (AECB): I would first like to comment on Mr. Patterson's paper. I suggest that all errors are human errors, whether they be in operation or back further in commissioning, manufacturing, or design. About all we can do is change the location of this error from the operator back to the designer, where hopefully more time can be given to considerations.

My question is in connection with Patterson's use (or nonuse) of repair time in computing reliability. I think he used the word "disallow." I take it that this means he

ignores repair time. Personally I cannot see how one can do this and come up with an honest evaluation of reliability of protective systems.

PATTERSON: When I said "disallowed," I meant just that, because in my mind repair time is nothing more than a gimmick. It is something that we pull out of the air. One might say: "Well, I have a module. I can pull it out in 10 seconds and replace it. Therefore, my repair time is 10 seconds." This would be true if a spare module were sitting there. What does one do when a component like an ion chamber, which is completely unaccessible except at shutdown, fails? What is the repair time for something of this sort? I can say that it is a matter of 10 hours. But, if the component fails at the beginning of the year's cycle, and I do not need it because of redundant channels, then the repair time for this particular device becomes fictitious. I think that we arrive at a much more honest, direct, and understandable figure by ignoring repair time in the calculations.

HAKE (AECL): There seems to be some confusion concerning the words "reliability" and "availability." If Mr. Patterson would look for results of the SCORE program, he would see how important are the considerations of repair time and replacement of the components.

GRIFFIN (AI): Along the lines of this last comment, when one has redundant circuits or coincident circuits and the repair time is ignored or the repair time is large, then the failure rates would be changed and a false security would be associated with the reliability of the system.

EPLER (ORNL): I note the absence of any comment as to the state of the system during the time of repair. If the system is left in its original state, then it is in a state of 2-out-of-2 coincidence in an initial 3-channel system. In this case there is a low order of protection. If, on the other hand, the channel is deliberately tripped to a safe failure prior to repair, then one has a 1-out-of-2 system and higher reliability, more comparable to the original 2-out-of-3 layout. If action is taken, then the question of repair time is of no consequence.

HAKE: The objective is to protect the people from hazard. If one quantitatively expresses the risk one is willing to take, then one can put some numbers on the time during which the system could be allowed to be down for testing and repair. If these numbers are not compatible with the risk one has decided to take, then the problem has not been solved, and one should not be in business.

PATTERSON: There is a tendency to say, after having worked these probabilities out, "Now my unavailability is so many minutes." People must understand that this unavailability must be viewed in a statistical sense because it is not known whether this unavailability is going to occur at the beginning or immediately following the test; one has no idea at what point one is going to lose the system. I am afraid that it leads to thinking in false security terms. That is why I keep "availability" and "reliability" completely separate.

JORDAN (ORNL): I want to raise a question concerning the figures of 10^{-5} or 10^{-4} presented by Mr. Patterson. We are coming close to the state of the art, and I want to raise the question: How do these figures compare with what has actually been observed at the MTR and ETR as Mr. Schroeder told us, or in the case of the HFIR? Do we find that a 3000-hour or a 5000-hour mean time between detecting unsafe silent failures is hard to achieve?

DEBOISBLANC: The MTR has operated continuously since 1951 and the ETR since 1956, and during this period of time there have been two classes of problems. One is the loss of reactor time in the absence of failure, and the other is the ability of the instruments to perform when there is really something wrong. I think that we can more readily agree on what is an allowable loss of reactor time or on the probability of malfunction which causes loss of reactor time than we can agree on what is an allowable probability of the instrumentation system as a whole to protect against an incident or a failure that would be embarrassing.

In terms of the loss of operating time, we have found that there is something serious enough that the instrumentation system should do something about it roughly once a year. The number of times in which the instrumentation systems have collectively failed to do this something is zero.

We have had cases in which the instrumentation system failed to protect properly against the rupture of cladding and the dispersal of fission products. But, even in these cases, some portion of the instrumentation told us that we had an effect and gave us enough time to shut down the reactor sufficiently to limit its consequences. This is more a commentary upon the system design than it is upon the total instrumentation ensemble performing as a reliability criterion. And that is a subtle point that must be considered. We can upgrade systems by making them more complex and more redundant, by looking at more variables, but we must make a careful distinction between when we have merely upgraded the reliability and when we have actually recovered operating time.

**Nuclear Methods of Malfunction
Diagnosis and Boiling Detection**

M. A. Schultz, Chairman

Survey of Boiling Detection Methods in Reactors

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To date, the detection of boiling in nuclear reactors has received relatively little attention but it presents a problem which has increasing importance as a result of the building of reactors like HFIR, and the projected building of large liquid metal cooled fast reactors for power production. This problem is, of course, not applicable to reactors in which boiling occurs as a designed operating condition, but only to reactors operating at as high a temperature as possible for high efficiency but in which boiling would give rise to potentially hazardous conditions.

It follows, therefore, that there is considerable incentive to develop a reliable and sensitive boiling detector.

Certain general observations regarding the possible methods for detection present themselves. In the most general terms, detection methods are based either on the receipt and recognition of signals from the phenomenon to be detected or are based on the interaction between the phenomenon and a "probe," using the term "probe" in its most general form.

As far as is known at present, the only signals emitted by boiling are acoustic signals.

The types of probe which may be used, in general, fall into those with a characteristic length small compared with that of the phenomenon to be detected and into those with a characteristic length large compared with that of the phenomenon to be measured. Efficient detection requires that the former of the two cases be used, since the effects of the phenomenon on the probe are much greater for these conditions.

When the characteristic length of the probe is large compared with that of the phenomenon, the effect caused by the phenomenon is in general small and efficiency is low.

In the case of bubble detection by "probes," we may list the following in two groups. Firstly, "probes" where the probe characteristic length is small compared with bubble size:

- 1) Optical methods
- 2) Ultrasonic methods
- 3) Electrical methods

Secondly, "probes" whose characteristic lengths are long compared with bubble size:

- 4) x- or γ -rays
- 5) Neutrons
- 6) Acoustical methods (see below for an explanation of the apparent anomaly)

<u>"Probes"</u>	<u>Relevant characteristic length</u>	<u>Relevant effects</u>
<u>A. "Probes" with characteristic length less than bubble size</u>		
1. Optical methods	Optical wavelength	Deflection; dispersion; change of path length
2. Ultrasonic methods	Wavelength	Deflection; change of path length
3. Electrical methods	Mean-free-path between collisions	Change in electrical resistance
<u>B. "Probes" with characteristic length greater than bubble size</u>		
4. x- or γ -rays	Absorption length	Change in absorption
5. Neutrons	Absorption mean-free-path Scattering mean-free-path	Change in absorption and/or in scattering
6. Acoustical methods	Absorption length	Change in absorption or scattering

It follows from this, that methods 1, 2 or 3 would be preferred, while methods 4, 5, and 6 would be of relatively little use. However, the above criteria are not complete. Considerable additional criteria or limitations may be imposed by the particular conditions under which the measurement or detection is to be made, viz.:

- a) the presence of a strong nuclear environment (neutrons, γ -rays, etc.)
- b) the presence of high temperatures
- c) the presence of high pressures
- d) the interposition of a pressure vessel and/or a biological shield between the region in which boiling will take place and a biologically safe region
- e) the presence, inside the reactor, of a corrosive medium
- f) the presence, inside the reactor, of many coolant channels, in any one or more of which boiling may occur.

Using these particular conditions, we may make an assessment of the a priori relative possibilities of the methods 1 through 6.

	<u>Method</u>	<u>a</u>	<u>b</u>	<u>c</u>	<u>d</u>	<u>e</u>	<u>f</u>
1.	Optical	B	D	E	B	C	B
2.	Ultrasonic	E	C	C	C	C	B
3.	Electrical	A	E	E	C	E	C
4.	x- or γ -rays	A	F	F	D-C	F	C
5.	Neutrons	X	F	F	D-C	F	D
6.	Acoustical	E	C	C	D-C	C	B

A $\rightarrow$ almost impossible
 B $\rightarrow$ very difficult
 C $\rightarrow$ difficult
 D $\rightarrow$ some effect
 E $\rightarrow$ small effect
 F $\rightarrow$ no effect
 X $\rightarrow$ essential to the measurements

We may conclude from this that the order of preference of the six methods due to these particular conditions is:

- 1) Neutrons
- 2) Ultrasonic
- 3) Acoustical
- 4) Optical
- 5) x- or γ -ray
- 6) Electrical

This order, combined with the order of preference due to characteristic length considerations, leads to a final order of preference of:

- 1) Ultrasonic
- 2) Neutrons

Acoustical probe and optical methods are probably very difficult to apply to a reactor due to the problems of insertion, transmission and detection against the effects of the environment and x- or γ -ray and electrical methods probably almost impossible to apply to a reactor owing to the environment.

We are therefore left to consider two "probe" methods, ultrasonic and neutron, the neutrons being present already in the reactor, and also to consider the use of acoustic emissions from bubbles, thermal noise and the fluctuations in a flowmeter caused by boiling. In these last methods, acoustic emission will have a characteristic length well-matched with bubble sizes since it is produced by the bubbles and also requires no "probe," while thermal noise and flowmeter fluctuations will probably have characteristic lengths large compared with bubble sizes and would not therefore be expected to be very efficient.

This analysis leaves us with five methods for the detection of boiling in reactors of which the first three have received the most attention. In this review, therefore, an account will be given of these three methods in some detail starting with the use of neutrons. The other two methods will be briefly mentioned subsequently.

Neutron-flux Measurements

The measurement of reactor kinetic characteristics by measurements of the neutron flux includes, among others, the method of applying to the reactor a

known input and of using the resultant neutron flux behaviour from which to deduce the reactor's kinetic behaviour. Among the known inputs applied by various investigators have been sinusoidal, delta-function, step-function, ramp-function, programmed binary sequences and inherent noise. Of these, all but the last must be applied from without the reactor and may therefore be awkward to apply through a pressure vessel. The inherent noise input depends on the random variation of neutron input to a reactor, ostensibly running at constant power, due to the statistics of neutron emission upon fission. The strength of this method lies in the inevitable presence in any reactor of this forcing function and in its wide-band characteristics. The presence of this forcing function removes the necessity of the application through a pressure vessel and the wide-band characteristics make it possible, in theory, to deal with a very wide range of frequencies simultaneously and not consecutively as in some of the other methods.

In essence, therefore, the methods consists of using a neutron detector and associated equipment to analyze the characteristics of the fluctuations of the neutron flux in the reactor, these fluctuations being due to the statistics of neutron emission on fission in the reactor, amplified and modified by the characteristics of the reactor [1]. If the neutron detector could be placed outside the reactor pressure vessel, a very favourable set of circumstances would exist in that no apparatus of any sort would be required within the pressure vessel and problems of penetration would be absent. However, Cohn [2] shows that unless the detector efficiency is at least 5×10^{-5} , measurements on neutron-flux fluctuations, or neutron-flux noise, will not yield the reactor characteristics. In most cases, to attain at least this value of detector efficiency, the detector must be placed in close proximity to the core and therefore pressure vessel penetration may well be required.

Assuming that the reactor's kinetic behaviour may be derived from an analysis of neutron-flux noise, the detection of boiling rests on the ability to observe changes in this kinetic behaviour due to the boiling. Noting that it is the onset of boiling which we wish to detect, and not the occurrence of vigorous boiling, it is possible to draw some general conclusions regarding the detection possibilities.

At the onset of boiling, the liquid will be, to some extent, subcooled and therefore boiling bubbles will persist only for a short time before collapsing. If it is assumed that the bubble formation and collapse rates are constant and equal, and that the bubbles are of uniform size, the total amount of bubble volume remains constant. If, in addition, the bubbles remained in the same region in the reactor, the reactivity introduced by the bubbles remains constant and a change in control rod position would restore criticality. The mean neutron-flux noise level would remain constant. However, if, as is very likely, the amount of bubble volume varies temporally about some mean value and/or the region of boiling does not remain stationary in the reactor, then the neutron-flux noise will be increased.

It would appear therefore that the effect which would render boiling detectable by neutron-flux noise methods is the fluctuation of total bubble volume with time, and/or changes in the location of the boiling, i.e., a second order effect. Furthermore, there appears to be no fundamental reason why these bubble fluctuations should occur at a definite frequency. It appears more probable that, given the range of bubble sizes which grow from different nucleation sites and given the irregular frequency of bubble appearance at each bubble site, the fluctuation of bubble volume and location with time will exhibit wide-band characteristics. Therefore it would be expected that the effect of boiling would be a general increase of neutron-flux noise over a wide frequency range.

This a priori expectation is in agreement with results given by Boyd [3] using electrically created boiling in a reactor. Colomb and Binford [4], using an electrical heater to produce boiling in a reactor, were able to produce detectable effects on the acoustic characteristics (see later) but the results of neutron-flux noise measurements were inconclusive. Tabor and Hurt [5] used neutron-flux noise methods to attempt to detect nucleate boiling in an orificed fuel element in ORR. They found that the onset of boiling gave violent oscillations of the neutron flux at a frequency of around 2 c/s. They estimated that about 170 cm³ of void was present or about 7.7 percent of the active region of the throttled fuel element consisted of void. They concluded from their measurements that bulk-boiling, "chugging" and possibly some flow reversal was occurring.

Fry, Roux, Ricker, Stephenson, Hanauer and Trinko [6] during a boiling test on ORR concluded that if nucleate boiling did occur, its presence was not observable in any of the spectra obtained and that the effect of nucleate boiling on the power-spectral-density curve, if any, was masked by fluctuations due to other causes. Schultz [7] found a similar lack of indication using CP-5, except that the normal reactor instrumentation gave some indications.

Stephenson, Roux, and Fry [8] consider reactivity, coolant-flow-rate, and coolant inlet temperature fluctuations as possible causes of the increase of noise level observed in a zero-power reactor, this increase being detrimental to the chances of observing boiling by this method. Yamada and Kage [9] also consider the effects of coolant-flow fluctuations on reactor noise. Both these groups of authors find it difficult to reach definite conclusions on these points.

Jordan [10] gives power-spectral density curves in which changes are attributed to boiling with a void fraction of 1.4 percent $\Delta v/v$ per element. A special fuel assembly, equipped with flowmeter and thermocouples, was orificed to reduce a downward flow of 1.8 ft/sec normally to 0.6 ft/sec to cause boiling at an estimated power of 2.2 Mw. The change in ion-chamber detected power-spectral density occurred at a reactor power of 2.35 Mw. The power-spectral density curve exhibited a multiple peak structure over the frequency range of 0.2 c/s to 2.5 c/s suggesting that the disturbance consisted of a set of pulses of fixed frequency, width, and amplitude. The frequency of occurrence was 0.33/sec and, with a bubble lifetime in the subcooled liquid of 0.2 sec, strongly suggests the type of behaviour noticed by Colomb and Binford [4].

In contrast to the above examples, Rajagopal and Gallagher [11] using the Saxton reactor show a very sharp and distinct peak in the neutron power-spectral-density curve at 16 c/s, the peak having a half-width of only about 2 c/s. The curve is shown only for the range of frequencies 10 to 20 c/s. The authors claim that investigation of the resonance at various operating conditions established that "it is definitely related to nucleate boiling." No details are given by these authors of the methods by which this relation is established.

Rajagopal [12] also gives noise resonance frequency characteristics for the Saxton, Yankee, and Trino reactors and states that the power-spectral-density of the neutron level fluctuations, as detected by an out-of-core ion chamber, in these plants has shown two well-defined resonances. Investigations of the resonances at various operating conditions establish that they are related to local boiling. Currently, no analytical formulation exists either for relating the resonance to local boiling or determining the absolute magnitude of local boiling.

These experiments, in which an attempt was made to establish that nucleate boiling in a reactor could be detected by neutron-flux measurements, can be

grouped into two groups: those in which the boiling is caused electrically and those in which nuclear heat is used to cause boiling. In the former case, the amount of electrical power needed to cause boiling can be measured during extra-reactor experiments and therefore the point at which boiling commences during in-reactor experiments can be known with some confidence. In those experiments, a general increase in noise at nearly all frequencies or no detectable increase in noise were noted.

In experiments using nuclear heat to cause boiling, the same degree of confidence in the prediction of the onset of boiling cannot be warranted. Furthermore, in most cases, no unambiguous alternative boiling indication method was included, usually due to the difficulty of providing such a method. It is not possible to state with certainty therefore in these cases that the observed effects or lack of effects were connected with nucleate boiling. In the case of the measurements of Rajagopal and Gallagher mentioned above, the occurrence of a distinct resonant frequency strongly suggests a mechanism other than boiling, although it is possible that this mechanism was excited by boiling if it occurred. In the absence of an alternative, unambiguous, confirmatory experimental technique, the conclusions must be imprecise.

This lack of precision arises, to some extent, from the nature of the experimental method. Normally, in any experiment, the parameters are varied one at a time, keeping the others constant, so that the effect of each can be studied. This admirable technique has not, so far, been followed in this field. All the parameters, whatever they are, which give rise to the observed power-spectral-density may be affected to an unknown amount by changes such as an increase in reactor power and to a large extent it will be a matter of chance whether an approximately true model is deduced from the observed spectra.

We conclude therefore that, so far, there is doubt whether the neutron-flux noise method is capable of detecting nucleate boiling in a power reactor and there is no evidence from which it is possible to predict the amount of boiling which could be detected in any particular reactor. To a large extent, this lack of detection appears to be due to the masking effect of the neutron-flux noise due to other causes not yet identified, especially at low frequencies. It would seem reasonable, therefore, to concentrate attention on the identification of these other causes so that the signal-to-noise ratio for boiling detection could be improved.

Ultrasonic Probe Measurements

The method of the ultrasonic detection of incipient boiling in essence consists of applying, to a probe in the liquid, ultrasonic energy of sufficient magnitude just to cause cavitation in the liquid at the surface of the probe. A measure of the amount of ultrasonic energy required to reach this condition enables certain conclusions to be drawn about the state of the liquid. The ultrasonic energy is introduced to the required region by means of a long probe which can penetrate a containment vessel through a force-insensitive mount. The interaction between the probe and the liquid causes energy to be reflected back through the probe and this energy can then be studied. Both the ultrasonic generator and the monitoring equipment can be remotely located outside the radiation shielding. The onset of cavitation causes the amount and character of the "reflected" energy to be changed and this change can be used to detect this onset. By using a frequency analyzer, the spectral characteristics of the "reflected" energy may be displayed on a cathode-ray tube. At low values of applied ultrasonic power, the spectrum consists of the fundamental frequency of the applied ultrasonic power and its harmonics. As the applied power is

increased, a point is reached where cavitation noise appears on the CRT trace as a band or bands of frequencies, easily distinguishable from the fundamental frequency and harmonics of the applied ultrasonic power.

Aerojet Products Inc. have applied this technique to experiments with the object of developing a practical incipient boiling detector in liquid metals [13 - 16]. These experiments show that the ultrasonic cavitation threshold power varies rapidly with temperature, both in water and in liquid metals. It also varies, as would be expected, with the local pressure at the point at which cavitation is being introduced. It has been shown that the techniques for introducing the ultrasonic probe and for detecting the resulting cavitation from without the liquid metal container exist and that it would therefore appear to be feasible to apply this technique to a liquid-metal cooled reactor.

This technique is the only one so far suggested which enables incipient boiling to be detected, using "incipient" in the sense of "about to begin, but not yet begun." From this point of view, it is the most desirable method in that it gives warning of an approach to a boiling condition and therefore gives time for corrective action to be taken. However, it has a very serious drawback in that the method gives information about the state of the liquid only in the very small volume in the immediate vicinity of the end of the probe. No information can be derived about the state of the liquid elsewhere in the reactor. Since a liquid-metal cooled fast reactor will presumably be operated at an average temperature such that boiling should not occur anywhere in the reactor, then boiling will occur only through local variations giving rise to local departures of temperature and/or pressure from the expected values.

These local variations in temperature and pressure may be due to manufacturing tolerances, distortions caused, among other things, by temperature, radiation or the like, differences of surface finish, corrosion, and the deposition or removal of material onto or from a surface and since many of these effects will change during the operating lifetime of the reactor, a sufficient knowledge of the magnitudes of these effects at all places and at all times to be able to predict the place at which boiling will first occur implies a detailed knowledge of reactor behaviour which should render boiling detection unnecessary. In the absence of such detailed knowledge, there is no means of deciding in which position to place the ultrasonic probe.

Another difficulty which may stand in the way of using this type of system in an operating reactor system is the generation of cavitation noise by the pumps which force the liquid metal around the system. Even in electromagnetic pumps, cavitation can occur [24] and with mechanical pumps, cavitation is easily possible. Since cavitation noise covers a very wide frequency range (see below), the onset of ultrasonically induced cavitation at the end of a probe must be detected above the background of possible pump cavitation. The relative amplitudes of these two effects cannot at present be easily estimated.

Acoustic Emission Measurements

A method for the possible detection of boiling in a nuclear reactor which does not involve the insertion of a "probe" into the reactor is that of listening for the acoustic emissions of the boiling bubbles. In this context, acoustic emissions are to be taken to refer to mechanical waves covering a possible very wide range of frequencies and not just waves whose frequencies fall within the audible range.

Acoustical waves may be emitted by bubbles through more than one mechanism. Boiling bubbles which detach themselves from a surface during nucleate boiling

will tend to undergo volume oscillations and in so doing will emit acoustic radiations and also boiling bubbles in a sub-cooled liquid will eventually collapse and, on collapse, the bubble will tend to emit an intense, short-duration pulse of acoustic radiation. It follows, therefore, that the spectrum of boiling noise will be a sum of the spectra caused by these two effects.

Minnaert [17] shows that the frequency of radiation emitted by a boiling bubble, radius R_0 , on detachment is given by

$$f_0 = (3\gamma P_0/\rho)^{1/2}/2\pi R_0,$$

where P_0 is the static pressure, γ is the ratio of the specific heats of the gas in the bubble, and ρ is the density of the liquid.

Given, therefore that the boiling bubbles exhibit a range of radii on detachment, due to local variations of surface conditions, then a broad frequency range of emission would be expected.

Several papers have been published on the sound radiated by the collapse of a vaporous cavity of which that by M. Harrison [18] is typical. In this paper, Harrison states that the pressure pulse measured experimentally can be fitted to the curve $p(t) = p_0 \exp(-t/\alpha)$, where α is in the range 10 to 30 μ s for water. Harrison further shows that a 10 μ s bubble should have a flat spectrum up to about 16 kc/s.

Work by Osborne and Holland [19] showed that, for boiling from an electrically-heated wire in water, the frequency distribution of the sound has a broad irregular maximum between 1 and 30 kc/s. This work suggests that the volume oscillation emission is small compared with the emission on collapse of the bubble. The spectrum which would result from a series of quasi- δ -function emissions of random amplitude and spaced randomly in time would be an approximation to wide-band white noise (Il'ichev, V. I. and Lesunovskii, V. P., [20]) and such a spectrum, modified by detector resonances and by system resonances or acoustical anisotropy, would explain many of the experimental results obtained by many investigators of the acoustic emission caused by boiling. A similar conclusion was reached by Saxe and Lau [21] regarding the spectrum of sound emission by cavitation caused in reactor fuel elements by the rapid flow of water in the coolant channels and was confirmed by the reproduction of the "cavitation acoustical" spectrum by dropping a stream of sand onto the microphone to cause a sequence of quasi- δ -functions of varying amplitude and spaced at random intervals of time.

From this, it follows that, in all probability, the acoustic emission from boiling consists of a quasi broad-band, white noise spectrum with possibly a "peak" at the high frequency end. It is not at all certain however that previous investigators have reached the high frequency end of the spectrum and there is little guidance from theoretical considerations on the approximate frequency of this. The above discussion has been centred on the results obtained using water as the fluid. The amount of reported work using liquid metals as the fluid is smaller, but nevertheless shows very similar behaviour in this fluid.

The problem of using the acoustic emission from boiling bubbles to detect boiling in a nuclear reactor therefore centres round the following points:

- 1) the need to be able to detect a quasi broad-band white noise emission whose upper frequency cut-off is not known.

- 2) the need to be able to detect this emission in the presence of possible cavitation noise which will itself also be a quasi broad-band, white noise spectrum of unknown upper cut-off frequency.
- 3) the need to be able to detect this emission no matter where it occurs in the reactor.
- 4) the need to be able to detect as small an amount of boiling as possible and as rapidly as possible.

Some work has already been done in this field of study.

Binford and Colomb [4] working with an electrical heater in ORR were able to detect boiling noise. They used an underwater microphone, the output from which was analyzed by a Hewlett-Packard wave analyzer with a 240 second integration time constant. Spectra are given within a frequency range of 0.5 kc/s to 4 kc/s and show several prominent peaks. Calculations indicated that boiling should commence on the heater at about 1 kW of electrical power. The amplitude of the acoustical noise was observed to be constant for heater powers below about 0.9 kW but to rise linearly at powers above this. The boiling noise, although detectable, was apparently small compared with the background "hydraulic" noise in the reactor.

A. J. Walton [22] performed experiments using lead-barium-titanate transducers and a Bruel and Kjaer frequency analyzer to measure the noise produced by boiling. He found that wide-band noise is produced when bubbles from a heated surface collapse on reaching a sub-cooled region. Also, gas bubbles which are produced by boiling in water containing air or in sodium containing argon act as resonators at frequencies inversely proportional to their size. He concludes that the resultant noise spectrum covers audio frequency range but that the shape depends on the number and size of gas bubbles present. Walton decides that the detection of boiling by "listening" is feasible provided that the noise produced by boiling can be distinguished from the background.

The work at Dounreay [23, 24] has been directed primarily towards the detection of nucleate boiling in liquid metal cooled reactors. The detection has been made by conventional piezo-electric transducers mounted on to the end of a mechanical waveguide, the end of which remote from the detector is located within the liquid metal. By this means, the detector is not operating under high temperature or high-radiation conditions. Typical spectra for boiling in liquid metal show similar characteristics, i.e., a broad spectrum of emission covering the range 600 c/s to 6000 c/s for water (limit of record) and 2 kc/s to 20 kc/s for boiling sodium (limit of record). The noise emitted by the flow of the liquid sodium without boiling shows approximately the same spectral shape as does the boiling of liquid sodium. The authors (Barclay, F. J. and Ledwidge, T. J. [24]) state that large amounts of sonic energy are available for detection, but that the main problem appears to be the uncertain background against which the boiling signal must be detected.

A variant of this approach is being investigated by Anderson, T. T., of ANL, who is endeavouring to develop a high-temperature piezo-electric hydrophone capable of being immersed in liquid sodium and yet still retaining a broad frequency response and fairly high sensitivity. Lithium niobate has been selected for the piezo-electric material to be used mainly as a result of its high Curie temperature (1210°C). The use of a detector capable of immersion in liquid sodium would remove the necessity of introducing a mechanical waveguide.

Dixon at Battelle NW is working on an electrostatic transducer also capable of being introduced directly into liquid sodium and capable also of withstanding a high value of nuclear radiation. This is an alternative approach to achieve

the same result as that of a detector using lithium niobate, about whose nuclear radiation sensitivity little is known. At present, the work is in too early a stage to be able to state with certainty whether either detector will function satisfactorily or to estimate relative performances.

The attempt to develop an immersed detector appears to have as its main appeal the removal of the necessity of using a mechanical waveguide. However, it is not at all obvious that this will lead to any improvement. The reduction in signal which may result from the use of a waveguide may be offset against the loss of gain occasioned by the need to use a crystal capable of operating under peculiar conditions. Also, the distortion in spectrum caused by the use of a waveguide may be very unimportant when the signal to be detected has white noise characteristics. Finally, the problems caused by penetration of a containment vessel by a mechanical waveguide appear to be comparable with those caused by similar penetration by an immersed detector.

In practice, the relative characteristics of various detectors, with or without waveguides, may well prove to be relatively unimportant. The important quantity may well be the amplitude of the boiling noise compared with that of the background noise. If the background noise is mainly caused by cavitation, then its spectrum and that of boiling noise will, judging by present evidence, be the same over most of the frequency range. It follows therefore that the discrimination may not be done on a frequency basis but must be done on a pure amplitude basis. It is possible that the spectra of the two sources could differ at the extreme upper end of the frequency range, allowing frequency discrimination to be used; but this region has not yet been explored.

The effects of the geometry of the reactor system on the signal received at any point due to a noise generation at any other point are being investigated by Anderson. The position for detection may well prove to be of importance, but the results will presumably be applicable both to an immersed detector and to a waveguide.

To summarise the situation of the detection of boiling by acoustic emission, therefore, it may be said that the method is feasible. The spectrum of emission from boiling in water and in liquid metals has been shown to cover a wide frequency range, the upper limit of which is unknown. The noise emitted by cavitation has been shown also to be generated by a similar mechanism and to have the same form of spectrum. It is expected that cavitation may occur in liquid-metal pumps, even electromagnetic type pumps [24, 25], and the detection of boiling may therefore rest on an ability to detect an increase of background noise. Work has been performed on the DFR at Dounreay using a mechanical waveguide and a normal piezo-electric detector and it has been shown that boiling can be detected, albeit against a relatively quiet background. Work is also being conducted at ANL and at Battelle NW to develop immersible detectors to eliminate the need for waveguides. So far, no work appears to be in progress to determine the upper frequency limit for boiling noise and for cavitation noise.

Other Methods

Other techniques reported in the literature include:

- 1) thermal noise (temperature fluctuations) in the coolant stream. Barclay and Ledwidge [24] are using mineral-insulated chromel-alumel thermocouples sheathed in stainless steel to measure the temperature fluctuations on exit from the flow channel. The noise bandwidth is limited to frequencies below about

1 c/s and to increase this, work is underway on the development of an eddy current probe. The authors state that the difficulty is to provide electrical insulation adequate for high temperature in a radiation field. Work on this method is also being started by Mulcahey at ANL.

2) Misch [26] measured the time between the initiation of an electrical heating step and the commencement of boiling from a wire as a measure of the state of the liquid. He used high speed photography and acoustic detection to determine the instant of appearance of the first bubble. In principle, this method enables the approach to incipient boiling to be studied but it suffers from the same disadvantage as does the ultrasonic incipient boiling detector in that it gives information about conditions only in a very localised region.

3) Investigators at Atomics International [27] measured the power-spectral-density of the fluctuations in a flowmeter in liquid sodium. An accelerometer was used to monitor the onset of boiling from the acoustical signals. Three curves of noise from the flowmeter are shown for conditions of no boiling, boiling in the test section and for boiling such that bubbles passed through the flowmeter. It is claimed that the shape of the power-spectral density curve over the range of frequencies from a few cycles per second to 100 or more cycles per second enables these three conditions to be distinguished from each other.

A possible disadvantage of the thermal noise and flow noise techniques is that, in a large power reactor, considerable background thermal and flow noise would probably occur in the absence of boiling and the signal-to-noise ratio may be unfavourable.

Conclusions

At present, it appears evident that three main methods are being investigated to produce an effective boiling detector for nuclear reactors--neutron-flux noise, ultrasonic probe, and acoustic emission detector. All three methods have one thing in common in that the detection will depend on the ability to detect wanted signals above an unwanted background.

In other respects, the three methods differ. The neutron-flux noise method utilizes what would appear to be a second-order effect, *i.e.*, the fluctuations in neutron flux caused by fluctuations in total bubble volume. The advantage of this method is that it requires no input to the reactor and can therefore, in theory, be applied to any operating reactor. The disadvantages would appear to be: 1) the possible difficulty of attaining the required detector efficiency in practice, 2) the insensitivity caused by the reliance on a second-order effect, 3) the long time required to achieve sufficient accuracy to make the detection, 4) the possibility that certain parts of the reactor may not be sensitive to the presence of bubbles, 5) the lack of knowledge of the causes of the large background neutron-flux noise, 6) the "low-pass filter" characteristics of the nuclear reactor.

The ultrasonic probe method utilizes the generation of cavitation by ultrasonic means to determine the state of the liquid. The advantage of this method is that, alone of the three, it enables the approach to a state of boiling to be determined, at least in principle. Its overwhelming disadvantage lies in the localization of its measurement to within a very small volume, and its insensitivity to the state of the remainder of the reactor. It may also suffer from a further disadvantage--an inability to detect the ultrasonically induced cavitation in the presence of cavitation due to other causes.

The acoustic emission method depends on the ability to detect the acoustic emission from boiling bubbles. Its advantages are that it requires no interference with the reactor beyond the possible insertion of a small detector or mechanical waveguide and that it relies on an effect directly associated with boiling bubbles and with either the number or volume of the bubbles. Its disadvantage is that, in all probability, the boiling noise will need to be detected in the presence of similar noise from cavitation.

Of these methods, only the ultrasonic method is capable of detecting incipient boiling conditions. The other methods are capable only of detecting boiling after it has started. However, the ultrasonic method is not capable of giving information about all points in the reactor; the acoustic method alone, in principle, is capable of detecting boiling anywhere in the reactor.

All three methods will depend on the detection of boiling signals in the presence of background signals of similar spectral form unless a) cavitation can be eliminated from nuclear reactors or b) the high frequency spectrum of boiling can be shown to differ from that of cavitation.

From this analysis, it may be concluded that the most important points to be clarified are:

- 1) the origins of the background neutron flux noise in power reactors so that their reduction or elimination may be considered.
- 2) the origins of the background acoustical noise in power reactors (most probably cavitation) so that their reduction or elimination may be considered.
- 3) the characteristics of the higher frequency parts of the spectra of boiling noise and of cavitation.
- 4) the relative sensitivities of the several detection methods under "quiet" conditions.
- 5) the relative backgrounds for the various methods in typical power reactors.
- 6) the relative speeds with which threshold amounts of boiling may be unambiguously signalled by the various methods in a typical power reactor and the consequent degrees of hazard to the reactor likely to be caused subsequent to onset of boiling before remedial action can be initiated and become effective.

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Discussion on Paper by Saxe

DEBOISBLANC (IDAHO NUCLEAR): Professor Saxe made a statement to the effect that there were no systems in which sensitivity had been determined. In the MTR we have for years used the neutron flux signal to detect the onset of boiling, and we have indeed been able to operate on the threshold of incipient boiling with a degree of sensitivity that allows us to approach within 1% of the boiling condition and to back off from this and completely eliminate it.

However, rather than seeing a continuous distribution of white noise, as you said you saw, we noticed that there was apparently, in some interval of time related to the velocity of the formation of the bubbles, a concordant formation of a series of bubbles. Likewise, in a characteristic time related to the average time for a bubble to collapse, there was a corresponding collapse. When the system is in a state of boiling, this gives a collection of negative-going pulses followed by a reversal of sign which is caused by the collapse of the bubbles. When the bubbles form, they do not grow in amplitude, but rather they grow in frequency of occurrence; thus it is possible, for example, to see one pulse in 10 seconds. When boiling increases, there is a gradual crowding in of these little spikes on the system, and they are not random noise. So a broad-band system would consider this to be a fairly high frequency sound, and it is very close to what Binford and Colomb noticed at the ORR in the neighborhood of 1000 to 5000 cycles/second.

This boiling detection is possible because the MTR has a constant void coefficient over the entire core volume. However, in the ETR we have a wide spatial variation in void coefficient (in some regions it is almost positive), and the point that you mentioned is very definitely true; i.e., there is a 20:1 difference in sensitivity for detection of boiling in the ETR as in the MTR. Therefore, in certain reactors, it is a fine thing to use the neutron-detection system as a boiling monitor. In others, it is totally worthless.

BALDERSTON (BOEING): Have you done a considerable amount of work on what causes boiling? Are there completely different approaches to your incipient failure detection? Could you, for example, prevent the boiling from occurring by looking at the causes and physics of boiling?

SAXE (N. C. STATE U.): Well, considerable work has been done on the physics of boiling. Certainly in water it is not fully understood, and in liquid metals the degree of understanding is considerably less. The problem is that presumably one wants to operate a power reactor as close to boiling as possible to increase the efficiency, but, in the case of the liquid-metal reactors, not to boil it. If it does start to boil, the boiling will be due to some local irregularity, and what will cause this is anybody's guess. So the physics of the matter is rather complicated.

BALDERSTON: In other words, you are saying that the heat, even under normal conditions, can vary sufficiently and still be undetectable at another location.

SAXE: Yes. If the reactor is being operated in such a condition that it should not boil anywhere, and if it does start to boil somewhere, the boiling will be due to some local irregularity.

BALDERSTON: When you say "local," do you mean in the fluid? Or do you mean outside? If it is somewhere else, why not detect there?

SAXE: Well, it will be somewhere in one of the fuel channels in the reactor.

BALDERSTON: Why not detect right at that location?

SAXE: This is a problem. How would one detect down in the fuel channel in the reactor? We do not know how.

LEGLER (AEC): There is a very intimate connection between the incipient boiling condition and the amount of superheat. Furthermore, the amount of superheat that can be tied up in a channel depends to a great extent on the surface condition of the channel that the fluid is flowing in. Now, it is highly desirable not to have any superheat tied up in the coolant because it can come out all at once if it is triggered. Some of the detectors which Saxe described, particularly the ultrasonic probe, can be used to test how much superheat one has at the fuel-channel surface.

BALLARD (SANDIA): You mentioned electrical measurements, but then you did not say anything about them again. Would not electrical measurements between fixed probes somewhere in the coolant path be effective in detecting the presence of bubbles as the coolant passes through the probes?

SAXE: I think the problem there is the medium in which one is working. There are small coolant channels between metallic fuel elements, and I doubt that any electrical difference could be detected if a void were put in.

BALLARD: I gathered that these probes would be put in the coolant loop and not down in the channel. Your probes would be placed somewhere in an accessible area, would they not?

SAXE: No. The boiling will occur in the fuel channel inside the reactor, and this is where it must be detected or one would not be able to detect it at all.

BALLARD: The bubbles, then, do not carry through the loop any farther than in the reactor area?

SAXE: In subcooled liquid, no.

BALLARD: I would assume that the cavitation bubbles, which you stated were a problem to acoustic boiling detection, would also be a problem to any of the other techniques.

SAXE: They certainly do not help!

SCHULTZ: I wonder, Dr. Rajagopal, if you want to defend yourself? Saxe was casting aspersions on your previously reported claim of boiling detection with a spectral-density peak occurring at 16 cycles.

RAJAGOPAL (WESTINGHOUSE): We never claimed that this peak was directly related to boiling phenomena. We have always thought that the resonance is caused by some phenomenon associated with boiling, but not directly by boiling. We do not understand what causes it.

SAXE: I think that this is true. The main trouble is that one cannot get in the reactor core and apply the eyeball test to see if it is boiling or to see what is happening; thus one has to use indirect methods.

Experience with an On-Line Reactivity Balance Calculation as a Guide for Reactor Operation*

J. R. ENGEL

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The Molten Salt Reactor Experiment (MSRE) is a fluid-fueled reactor that uses a mixture of molten fluoride salts as the circulating fluid. The system operates at about 1200°F and near atmospheric pressure with a maximum power of 7.2 Mw. A comprehensive description of the reactor system is available in the safety analysis report.¹ The reactor was first made critical in June 1965 and since that time it has accumulated over 6000 equivalent full-power hours of operation.

During reactor operation a reactivity balance² is automatically calculated every 5 minutes by an on-line digital computer. The computer announces to the reactor operators any results that may be significant from the standpoint of orderly operation of the reactor. The balance is an algebraic sum of all expected reactivity changes from a reference condition established at the start of the first significant power operation. The appearance of any residual reactivity (either positive or negative) indicates either inadequate calculation of the magnitude of expected reactivity effects or the appearance of other effects that are not included in the calculation. Thus it is necessary to first establish the adequacy of the calculation for describing normal reactor behavior before it can be used to look for abnormalities.

During the early stages of power operation the reactivity balance was used as the primary tool for evaluating the xenon poisoning in the MSRE. In this application, it was assumed that all other reactivity terms were calculated

*Research sponsored by the U. S. Atomic Energy Commission under contract with the Union Carbide Corporation.

correctly and that any residual reactivity was associated with ^{135}Xe . This analysis gave a steady-state xenon poisoning of 0.35% $\delta k/k$ at 7.2 Mw; the anticipated poisoning had been about 1%. The lower value was confirmed by samples of the reactor offgas in which the isotopic ratio of ^{134}Xe to ^{136}Xe was measured to determine the ^{135}Xe poisoning. Analysis of the xenon reactivity transients associated with power changes was then used to evaluate parameters in developing a mathematical model to describe the xenon behavior in this reactor. Where possible, independent measurements were again used to confirm or lend support to values derived from the reactivity behavior.

Figure 1 shows the results of reactivity balance calculations at zero power with no xenon present for the entire power history of the reactor (through August 1967). This indicates good representation of all the long-term reactivity effects. This figure also illustrates the relationship between reactivity balance results and the reactor operating limits at $\pm 0.5\%$ $\delta k/k$.

After the accuracy of the reactivity balance had been established, it was reasonable to use the results to monitor the reactor performance on-line. Figure 2 shows the results of on-line calculations during a recent period of MSRE power operation. (The reference condition for this figure is the start of power operation in December 1965.) Only a small fraction of the data is presented, but each point is the result of an individual calculation. Thus, the point-to-point scatter is representative of that encountered in operation. In terms of orderly operation of the reactor, the reactivity balance is required to indicate changes in the range of 0.1 to 0.2% $\delta k/k$ since rapid changes of this magnitude will not adversely affect the reactor. In this respect, Fig. 2 (along with Fig. 1) illustrates that there has been no anomalous behavior of the reactor.

Detailed examination of Fig. 2 shows small changes in reactivity in the range of 0.02 to 0.03% $\delta k/k$. These changes are too small to have any safety significance but they are outside the normal scatter of results and are, therefore, real. In all cases the changes can be associated with specific changes in reactor (or computer) operating parameters. However, we have not yet, in all cases, been able to identify the mechanisms through which the changes in operating parameters affect the reactivity.

Results from the current operation of the reactor indicate a small change in the steady-state ^{135}Xe poisoning in the reactor. This suggests a change in one or more of the inherent parameters that govern xenon behavior. We have obtained some support for this from measurements of the reactor power spectral density but no direct correlations have been developed yet. The study of this and other minor effects is continuing. Thus, in addition to providing an on-line monitor for normal operation, the reactivity balance has proved to be a useful tool for analyzing detailed aspects of the operation that might otherwise not be apparent.

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Figure 1

Long-Term Drift in Residual Reactivity in MSRE

Figure 2

Residual Reactivity During MSRE Run 12

Reactivity Anomaly Detection in the HFIR with an On-Line Computer*

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INTRODUCTION

The increasing complexity of nuclear reactor control problems and the growing economic incentive to operate reactors as closely as possible to the design limits have necessitated serious consideration of fully automating reactor control systems. However, control system designers have been unable to reliably close the loop and automatically withdraw control rods at the request of the flux (or power) controller without a check on the validity of the reactivity request. As a result, the reactor operator is vested with the responsibility of performing or approving a rod withdrawal operation in response to a flux controller demand. He is required to assess whether a request for reactivity addition by the servo system is a reasonable one or whether in fact there exists some reactivity anomaly indicative of a dangerous situation. In most cases, the data processing required to decide whether or not a reactivity anomaly exists is beyond the capability of an operator who is not supplied with computational assistance.

In the case of the High Flux Isotope Reactor (HFIR), large and rapidly changing ^{135}Xe poisoning effects are a complicating feature. During a transient which follows a decrease in reactor power as a result of a legitimate reactor scram, an accidental control-rod drop, an ac power failure, or the partial loss of coolant pumps, it is virtually impossible for the operator to evaluate the "normal" reactivity requirements because of the rapidly changing xenon concentration. When

*Research sponsored by the U. S. Atomic Energy Commission under contract with the Union Carbide Corporation.

such a decrease in reactor power occurs, it is desirable to return to full power as soon as possible, while at the same time being able to recognize an anomaly which may have been the unrecognized cause of the shutdown. If the reactor cannot be returned quickly to full power, an unnecessary shutdown can occur as a result of xenon poisoning. This, in turn, can result in the premature loss of a core due to samarium poisoning, inasmuch as partial refueling is not possible in the HFIR.

REACTIVITY CHARACTERISTICS OF THE HFIR

The estimated total reactivity change in a typical 21-day cycle of HFIR operation is \$13.00. The rate of change during the initial ^{135}Xe buildup is \$0.40/hr, averaged over the first 12 hours of operation. From Fig. 1, it may be noted that 20 rod adjustments (shimming operations)

Fig. 1. HFIR Rod Position vs Time After Start-Up.

are required during this period. This rate is adequately slow, and the repeatability from core to core is sufficient to permit manual evaluation of the reactivity demand. The operator can compare the rod positions at any time with the values from the previous cycles and thus withhold rod movement whenever the demand appears to be excessive. However, during transients following a power reduction, the reactivity demand is about \$12.00/hr, or 30 times the normal maximum after startup. The exact rate is a function of the age of the core, and the variation may approach \$3.00/hr over the core life. The total reactivity addition required will depend on the duration of the reduced power operation.

Typical requirements are \$2.00 to \$3.00; however, a maximum of \$10.00 to \$12.00 is possible.

Considering the magnitude of reactivity change that is possible and the fact that the rate is variable over the core cycle, it is not difficult to conclude that a significant reactivity anomaly could develop without being readily detected by the operator.

To aid the operator in detecting and preventing reactivity anomalies in the HFIR, an on-line Control Data-1700 digital computer has been acquired to provide adequate control information. Continuously measured heat power and temperature information from the existing instrumentation will be used to calculate the expected reactivity balance at all times during the life of a core for both normal operation and transient conditions. By use of an appropriate reactivity model for the system, a continuous calculation will be made for changes in the reactivity associated with fuel burnup, burnable poison depletion, fission products, and temperature changes.

REACTIVITY MODEL

A basic requirement for an on-line reactivity model is simplicity. To ensure a high speed of response and a minimum computer memory, a one-group point reactor model was selected for the initial evaluation. Refinements and increased complexity will be added only if necessary to obtain the required accuracy. The cross sections used before reactor data became available were obtained by a least-squares fitting to solutions from a multigroup, multiregion code used for the original design work. The reactivity model has been described in detail in another report;¹ therefore, only the salient features are summarized below.

The major contributions to the reactivity change are given by Eq. 1:

$$\Delta\rho_{\text{net}} = \Delta\rho_{\text{fuel}} + \Delta\rho_{\text{Boron}} + \Delta\rho_{\text{Xe}} + \Delta\rho_{\text{Sm}} + \Delta\rho_{\text{FP}} + \Delta\rho_{\text{rods}}. \quad (1)$$

The term $\Delta\rho_{\text{FP}}$ is a lumped fission-product term which is linear with burn up and has an effective absorption cross section of 190 barns. The $\Delta\rho_{\text{rods}}$ term is inferred from the measured rod positions and rod worth calibration data. The remaining terms are the fuel, boron, xenon, and samarium constituents; these can be described by conventional differential equations and solved simultaneously. The appropriate value of thermal neutron flux is obtained from the standard relationship relating power level and fission rate:

$$P = C_1 E_f \phi. \quad (2)$$

The system of equations for the HFIR has been programmed and solutions have been generated by an off-line computer to aid in final cross-section fitting. Data from several HFIR transients have been analyzed by using rod calibration curves to calculate reactivity from the measured rod positions. A reactor power reduction to 50 Mw after

7.8 days of operation at 100 Mw exhibited the transient shown in Fig. 2. These data were used in adjusting the ^{135}Xe cross section to the present value because the observed changes were predominately due to xenon. Figure 3 shows the ability of the model to track the steady-state operation. The initial changes are dominated by the burnable poison (^{10}B) and the ^{135}Xe buildup. The transient following the reduction to 10 Mw is the most frequently observed transient in the HFIR, because a majority of the postulated malfunctions result in a controlled power reduction to this level.

Fig. 2. HFIR Reactivity Transient Following a Power Reduction to 10 Mw.

From Figs. 2 and 3 it can be seen that the maximum discrepancy using the current model is less than \$0.50 for both transient and steady-state operation. Since the HFIR safety system can protect the core from step reactivity insertions of several times this amount, a \$0.50 error is acceptable. A detailed error analysis of the model cannot be made until an on-line rod calibration technique is perfected.

The remaining consideration for on-line applications is the speed of response of the calculation. Off-line calculations have shown that two iterations each minute are required to solve the differential equations without excessive truncation or round-off errors. Therefore an on-line computer must be able to solve the system of equations within 30 sec to keep pace with the process. The HFIR model is currently being adapted to the CDC-1700 computer for on-line evaluation. A preliminary run has demonstrated an average solution rate of 0.2 sec per iteration. The reactivity model therefore has the required characteristics for on-line detection of reactivity anomalies greater than \$0.50 in the HFIR.

Fig. 3. HFIR Reactivity Transient Following a Power Reduction After 7.8 days at 100 Mw.

MEASURED REACTIVITY CHANGES

The recognition of an anomaly clearly depends on the discrepancy between an accurate measurement of the reactivity and an accurate calculation of expected change. Thus accurate knowledge of the control-rod calibration is needed. A study directed toward developing a method for on-line determination of control rod worths during operation at routine power has been conducted by Fry².

He demonstrated that the technique is feasible for the HFIR application and made recommendations on how to implement the method for routine use. This approach is expected to improve the sensitivity of anomaly detection in the HFIR by improving the accuracy of the measured change in reactivity. The variations in rod worth are not accounted for in the present reactivity model and are a likely source of error.

FLUCTUATING REACTIVITY ANOMALIES

The reactivity anomalies previously discussed consist, in each case, of a net reactivity discrepancy which must be viewed as a potential reactor excursion. Abnormalities may also be present in the fluctuating reactivity, which is very small and has a zero average amplitude. This type of abnormality is detectable by analyzing the computed spectral density of the neutron fluctuations observed at the output of an ionization chamber. Basically, the analysis consists of comparing spectra

accumulated on a day-to-day basis with a spectrum characteristic of normal operation; gross deviations in the overall amplitude or the detailed spectrum shape, or both, indicate an anomalous reactor condition. Unlike the reactivity anomalies discussed previously, the anomalous power spectrum is not necessarily indicative of an incipient reactor excursion. However, it may be indicative of an incipient reactor system failure;* consequently, on-line real-time computation and monitoring of the neutron spectral density is highly desirable.

For the purpose of detecting all varieties of fluctuating reactivity anomalies, it is essential to have a flexible spectral density computational procedure. In general, it is not known in advance in which frequency region of the spectrum the peaks or valleys characterizing the anomaly will appear. Hence, it is necessary to have virtually continuous control of the filter center frequencies and bandwidths over several decades of frequency, as well as a selectable number of frequencies at which the spectral density is estimated.

Dynamic range is also an important requirement, since the magnitude of the spectral density may well change by three orders of magnitude over the frequency range being monitored.

The above requirements suggest the use of an on-line digital computer for performing the spectral density computations. The conventional analog spectrum computational methods which have been used at ORNL in the past³ do not seem appropriate because of the stringent requirements of continuously selectable frequencies and bandwidths coupled with real-time computation at a large number of frequency points. Until recently, the algorithms employed in digital spectrum computation were much too slow for real-time operation, but with the advent of new Fast Fourier Transform (FFT) algorithms,^{4,5} which are approximately 30 times faster than the previously available digital methods, this situation has changed. When implemented on a modern process computer having a machine cycle time of about 1 μ sec, the FFT algorithm can calculate a reactor spectrum consisting of 250 or more frequencies within 2 min following the recording of the fluctuating signal. The center frequency and bandwidth structure provided by the digital computation can be varied easily to suit the needs of the analysis, and dynamic range is almost unlimited. The number of effective digital filters obtained is limited only by machine-core storage availability.

The performance of the FFT algorithms as applied to spectra computation has been checked by digitizing neutron fluctuation signals from the HFIR and comparing the results with those obtained with the existing ten-frequency analog spectral density analyzer.³ The improved frequency resolution and flexibility afforded by the FFT digital approach revealed fine structure in some spectra which previously had gone undetected. For comparison, Fig. 4 illustrates the results of a digital spectrum computation and the present analog computational procedure. The outstanding features of the comparison are the two

*D. N. Fry, "Neutron Density Fluctuations as a Reactor Diagnostic Tool," these proceedings.

Fig. 4. Comparison of Digital and Analog Computation of an HFIR Neutron Fluctuation Spectrum.

structured resonances at approximately 2.5 and 28 Hz, which were undetected by the analog analysis because of a more limited number of analog filters having wider bandwidths.

The FFT algorithms are presently being adapted to the CDC-1700 computer for routine on-line spectrum computations at the HFIR. In addition to the monitoring function, this capability for reducing fluctuation data will permit rapid observation of changes in the spectrum resulting from various purposely induced system perturbations. By use of such a cause-and-effect approach, it is anticipated that an improved understanding of the mechanisms causing specific spectral anomalies can be obtained.

Acknowledgements

The author wishes to recognize the contributions of H. P. Danforth in preparing the reactivity model calculations and the computer programs for on-line use at the HFIR. The development work for the FFT method was done by R. C. Kryter, and his contributions to the section entitled "Fluctuating Reactivity Anomalies" are gratefully acknowledged.

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Discussion on Papers by Engel and Bullock

NIMS (ADPA): I would like to direct this question to both Mr. Bullock and Mr. Engel. During the initial operation and check-out of these reactivity-balance methods, what sort of confusion did you run into if you found that the reactivity did not exactly balance? What procedures do you follow at the present time when you get an indication of a reactivity anomaly outside your acceptable limits, which I think were ± 50 cents?

BULLOCK (ORNL): I should probably clarify my position. At the moment, the HFIR computer is not on line. The question of operational procedure in the event of an anomaly outside the dead band has not been resolved. I think it is going to be this old problem of falling back on the human: the operations supervisor is probably going to have to make an assessment. It will take a period of time for confidence to be gained in the computer system. As more experience is gained, the operators will probably rely more heavily on the computer calculations, and then they will not counter its judgment. The operators will probably not go in the unsafe direction.

ENGEL (ORNL): In connection with the MSRE, the computer was essentially operational from the hardware and software standpoints when we first went into power operation with the reactor; so we were able to use it from the beginning.

I indicated that we do make use of measured parameters whenever possible. This leaves us with only a few areas within which we might expect anomalies, or rather differences between the calculated and observed behavior. We feel that we have a good indication of many of the nuclear characteristics, even those that cannot be measured directly. In particular, we feel that we understand such things as samarium, which is one of the major contributors in the reactivity balance, burnup, and buildup of plutonium and other isotopes.

When we observed the xenon anomaly, which I mentioned in my paper, it had a rather characteristic time constant. It looked fairly apparent at the time that it was in fact xenon, and so the problem was approached from that viewpoint. There were, of course, other criteria for judging the behavior of the reactor; various samples removed from the reactor for chemical analysis all indicated perfectly normal behavior. We experienced steady behavior in all other respects, and so there was no reason to believe that this particular phenomenon was anything other than xenon, and this conclusion was later borne out.

With respect to what will be done when an anomaly occurs, I want to emphasize that the MSRE is a particularly slow moving system. We have not, in fact, experienced an out-of-tolerance situation as yet. However, I would expect that, if an anomaly were to develop, it would develop relatively slowly. There would be some sort of consultation between the operators and supervisors, and, if it were truly an unexplained phenomenon, the reactor would undoubtedly be set back to low power to investigate the phenomenon. We are absolutely committed to do this when the anomaly exceeds 0.5% in reactivity. I think that we would undoubtedly reduce power long before it reaches that point.

DEBOISBLANC (IDAHO NUCLEAR): Mr. Bullock, how do you avoid the problem of aliasing when applying the fast Fourier transform to a finite record length?

BULLOCK: Mr. Kryter, would you comment?

KRYTER (ORNL): Aliasing is not a problem if you sample the signal at a high enough rate. One must satisfy the criterion that one has sampled rapidly enough to correctly describe the original fluctuating signal. This is all well known; the sampling rate must be at least twice the highest frequency component that is available within the spectrum. Maybe you are talking about a different sort of aliasing?

DEBOISBLANC: That is exactly what aliasing is, but I was referring back to Mr. Bullock's description of the on-line application. For conventional data processing, what you say is true; because then you have a running integral and this avoids the aliasing problem. But I thought Bullock said that you used a "chopped" section of the record. How do you avoid aliasing there?

KRYTER: Each chopped section constitutes an individual record capable of yielding estimates of the power spectral density. All such estimates can be regarded as being statistically independent, and therefore one can have an arbitrary time break (i.e., a lack of sampling) between such blocks without influencing the results. When one block is processed, it furnishes an estimate of the power spectral density at a number of discrete frequencies—and it may be a very poor estimate in itself. But, when many of these blocks are ensemble averaged—and I mean by "many," several hundred—then a good statistical estimate is obtained.

COLE (ORNL): In line with the previous question regarding sampling procedure, it might be in order to give some idea of the length of sample used. Bullock mentioned the length of time it takes to compute a spectrum, but what sort of data sampling time do you have?

BULLOCK: This depends upon what frequency range you are interested in. Typically we use a sampling time of about 10 to 30 minutes. This may sound like a long time, but we are faced with this problem of having to record a lot of information to infer some property of the reactor because we are dealing with statistical phenomena.

Application of Noise Analysis Techniques to Detection of Incipient Malfunctions*

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ABSTRACT

Techniques are described which utilize noise or fluctuation signals from electronic transducers sensitive to vibrations, temperature, power, conductivity, pressure, sound level, flow rate, etc., to detect and study abnormal conditions in reactor systems or test models.

Applications which are summarized range from feasibility tests, to development of automatic on-line, incipient malfunction detection systems. Specific phenomena encountered in these applications include: boiling; cavitation; gas entrainment through a reactor core; movement of bubbles through pipes, hydraulic transients; mechanical vibrations; and breakdown of electrical insulators.

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INTRODUCTION

Noise analysis is an engineering specialty which includes the development and application of equipment and techniques for obtaining information from noise signals or noise contaminated signals. Statistical techniques, involving filtering processes in the frequency and time domain are used to discriminate against unwanted signals and thereby push measurement or detection sensitivity beyond conventional limits. Because they offer the highest possible sensitivity, noise analysis techniques are particularly applicable for diagnostic work and detection of incipient malfunctions.

The data for tests described in this paper were recorded using portable analog data recording systems similar to that shown in Figure 1. The data analysis and instrument development was carried out in the analog data reduction laboratory shown in Figure 2. The following sections describe how this equipment is utilized to develop simplified special purpose instrumentation for detecting incipient malfunctions in reactor systems and test models.

GENERAL METHODS

The basic experimental approach to the problem of detection of incipient malfunctions is summarized below:

- 1) Noise or fluctuation signals from existing or specially installed electronic transducers sensitive to vibrations, temperature, power, conductivity, pressure, sound level, flow rate, etc., are recorded on magnetic tape while the phenomena to be studied is artificially induced or simulated.
- 2) The recorded signals are reduced in terms of amplitude versus frequency versus time to determine time and frequency dependence of signal to noise ratio, and to determine optimum pass band and detection method.
- 3) A detection system is simulated and optimized using general purpose analog amplifiers, filters, detectors, comparators, and digital logic.
- 4) A special purpose unit (if required) is constructed with optimum features of prototype and tested with tape recorded signals.
- 5) The detector system is installed and tested under actual operating conditions.
- 6) The time response and sensitivity of the detector system is checked periodically by inducing or simulating the phenomena to be detected. A simulation system, if required, is installed as part of the detection system.

7) Annunciators, meters, chart recorders, oscilloscope displays, if required, are provided in test area. Signals from accelerometers, when used, are filtered to discriminate against background noise and fed to loudspeakers in test area.

Figure 1. Portable Analog Data Recording System

Figure 2. Data Acquisition and Reduction Facility

APPLICATIONS

Examples of recent work at Atomics International on incipient failure detection are described in the following sections. The first study involves feasibility tests of various detection techniques. The last two examples involve the development of simplified, on-line instruments for automatically detecting abnormal conditions and preventing damage to test apparatus. A summary of the sensors and detection methods used in these three examples is presented in Table 1.

DETECTION OF GAS IN LIQUID SODIUM

Experiments were conducted at the SRE⁽³⁾ during non-nuclear operation, to study the feasibility of using accelerometers, sensitive amplifiers and spectrum analysis techniques, to detect formation and collapse of voids and gas entrainment in sodium. The technique involved installing accelerometers on pipe brackets at a number of points along the primary sodium lines to the core and moderator. Gas entrainment was induced by intentionally permitting the

TABLE 1

APPLICATIONS OF NOISE ANALYSIS INSTRUMENTATION TECHNIQUES

Events	Problems	Sensors	Methods of Detection
<u>Boiling</u>			
In test section simulating LMFBR fuel element and flow channel	Burnout of electrical heater, partial channel blockage during transient boiling tests	EM flowmeter at inlet to test section Accelerometer mounted on test section Pressure — at outlet of test section	Rate of change of noise amplitude in 1-cps to 10-cps band Rate of change of integral of noise amplitude in 1-kc to 3-kc band Amplitude of noise in 1-cps to 10-cps band
Electrical breakdown of ceramic insulation in high heat flux heaters	Heaters burn in two, destroying evidence of failure mechanism and damage instrumented test section	Current Line Voltage	Rate of change of 60-cps heater input current and line voltage Rate of change of instantaneous product of input current and line voltage
<u>Void Movement</u>			
In pipes and flow channels at SRE	Reactivity perturbations as voids pass through core Hydraulic transients, gas entrapment in flow channels, pumps, etc., affects flow stability.	Accelerometers mounted along pipes, and on structures extending from core loading surface into the core such as heaters, pipes, and fuel hanger rods	Modulation of narrow band noise components in 200-cps to 10-kc band. (Maximum signals detected at natural frequencies of vessel, pipes, cavity, rods, or accelerometer supports.)

pump to draw in cover gas (helium). Recordings made before, during, and after gas was induced were analyzed to determine frequency regions that were most sensitive to the void effects. The signals were then passed through filters tuned to the optimum frequencies for the respective accelerometer locations, and displayed simultaneously on a multi-channel chart recorder in the form of rms level versus time. In this way it was possible to visually observe correlations between the various accelerometers.

Results of these experiments indicated that accelerometers apparently cannot be used to trace passage of particular gas voids through the piping system. However, gas entrainment and the effects of gas entrapment at local points, along the primary sodium lines to the core and moderator, can be detected from changes in amplitude and frequency content of accelerometer signals.

During these experiments, hydraulic pressure transients (water hammer) were detected by all accelerometers attached to sodium lines between the outlet of the intermediate heat exchanger and the inlet to the reactor core vessel. The transients occurred at random at the rate of about 1 to 3 per second, and were observed to propagate along the sodium lines to the core and moderator at about 5000 ft/sec. Using triangulation techniques, the transients were traced to their origin, a particular section of the sodium line to the moderator, where sodium flow was directed vertically downward for about 4 feet. The transients were attributed to voids being swept down out of a higher region of the moderator line against the force of gravity.

A second series of tests were conducted with accelerometers attached to core components at the loading face to determine if acoustic emissions could be detected from bubbles coming up through the reactor core. A pipe was installed through an empty fuel channel, to permit injection of known quantities of gas at the base of the core. One accelerometer was attached to the pipe, a second to a tubular heater, and a third was attached to a steel hanger rod. The heater and hanger rod were parallel to the bubble injection pipe at radial distances of one foot and four feet, respectively. A specially devised pressure regulator,⁽⁴⁾ was used to inject known quantities of gas into the core. In this way bubbles of known size could be induced and their associated acoustic emissions compared as they came up through the core region, the sodium pool above the core, and finally emerged at the upper surface of the sodium pool. Core viewing equipment was used to observe bubbles as they broke the surface. Manually operated switches were used to provide electronic indications of bubble injection (below the core) and the emergence of the bubbles at the surface of the sodium pool. Typical optimized response signals for bubbles of 0.5 liters and 0.05

liters are shown in Figures 3 and 4, respectively. Note that during the first two seconds after injection, while the bubbles were coming up through the core, they could only be detected by the accelerometer connected to the gas injection

Figure 3. Accelerometer Response to Acoustic Emission from 0.5-Liter Bubble

Figure 4. Accelerometer Response to Acoustic Emission from 0.05-Liter Bubble

tube. However, once the bubbles emerged from the core region and began to rise in the pool above the core, all accelerometers responded to acoustic emissions from the bubbles.

Results showed that accelerometers attached to core components at the loading face could be used to detect bubbles (down to at least 0.01 liter in size) rising from the core. Considerable information was gained from these tests, with regard to the capabilities (and limitations) of accelerometers for application to detection of voids in sodium systems.

DETECTION OF ELECTRICAL HEATER MALFUNCTIONS IN CLADDING MATERIAL TESTS

As part of a program to test cladding materials for LMFBR,⁽⁵⁾ 2-inch-long high flux graphite heaters were being operated at heat flux levels up to 1.0×10^6 Btu/hr-ft² at the 1/4-inch diameter (about 3.8 kw). This involved rms current levels around 240 amps through the 1/8-in. -diameter graphite element. Deterioration of the electrical connection to the graphite rod frequently led to overheating, arcing, and premature destruction of the heater.

A current transient detector system with six independent channels was designed and installed. The detector circuits were set to cut off heater power if the instantaneous current level exceeded a selected value for a fixed time limit or if the rate of change of current level exceeded a selected value for a fixed time limit. The rate circuit was used to detect electrical breakdown of the insulation during initial approach to power. The level trip circuits were used during On-Off power cycling tests.

Now, if arcing begins to occur at the connection to the graphite, the electrical noise trips the detector and power is removed before heater components are destroyed. After a simple mechanical adjustment of the contact electrode, the heater can usually be returned to operation. The application of this incipient failure detection system has permitted the extension of heater life sufficiently to meet test program requirements, i. e., induce cladding failure.

DETECTION OF BOILING AND HEATER MALFUNCTIONS IN TRANSIENT BOILING TESTS

As part of a Reactor Safety program,⁽⁶⁾ 15-inch-long, 1/4-inch-diameter electrical heaters were under development for thermal hydraulic simulation of LMFBR fuel pins. These heaters with a heat flux capability up to 10^6 Btu/hr-ft² (at 24-kw electrical input) were to be used in transient boiling tests to obtain data on superheating of liquid sodium in LMFBR fuel pin geometry. The tests

were to be conducted by establishing flow and subcooling and then increasing heater power until boiling was established.

The first problem was to develop a boiling detection system⁽⁷⁾ which could be used to protect the high heat flux heater from burnout due to vapor blanketing.

Tape recordings were made of flow fluctuations and mechanical vibrations in a radiantly heated test section from an EM flowmeter and an accelerometer, respectively, during transitional boiling conditions. Power spectral density analysis and time history analysis of selected spectral components indicated that both the flowmeter and accelerometer signals could be used to detect the onset of boiling in the sodium test loop. However, for the purpose of preventing heater burnout, the flowmeter signal had a superior signal to noise ratio, particularly at low flow rates where burnout conditions are more likely to occur. The main problem with the signal from the accelerometer was that any phenomena which produces wide band mechanical vibrations in the vicinity of the boiler vessel, such as acoustic emission from system components during thermal transients, turning a valve, rapping on the test chamber, etc., creates large amplitude signal bursts, which, on the basis of frequency content, cannot be readily distinguished from signals produced by boiling.

An electronic detection circuit was designed to analyze the EM flowmeter signals and provide an alarm signal at the onset of boiling. Signals characteristic of the onset of boiling from the EM flowmeter are amplified, filtered, and coupled to a rate- and level-sensing comparator which trips a relay, cutting off power to the high heat-flux heater after a selected time delay. This device permits the heater to operate without danger of burnout due to surface boiling at power levels comparable to, or in excess of, those of a fast reactor fuel element. If the heater were allowed to operate at full power during surface boiling conditions, the temperature would rise at rates approaching 1800°F per second. In previous operation under similar conditions, and without the boiling detector circuitry, heater failure usually occurred within one to five boiling tests. However, with the aid of the boiling detection system, more than 70 successful sodium boiling tests under various flow, power, and pressure conditions were carried out with one high flux heater; and more than 170 boiling tests have been completed with a second heater, which is still in operation.

An audio warning system was developed and installed in the test area. In this system, signals from an accelerometer are filtered to discriminate against background noise and are used to drive a loudspeaker in the testing area as an audible indication of possible boiling. On one occasion when the automatic boil-

ing detector, described above, was bypassed, the operator, in the course of changing test conditions, inadvertently applied enough power to the high flux heater to cause boiling. The boiling sounds coming over the loudspeaker system alerted him in time to manually cut off power and thereby prevent heater burnout.

The second problem was to detect sudden changes in heater electrical resistance which would indicate failure or electrical breakdown of the insulation material.⁽⁸⁾ This problem arose after the first heater finally gave out due to electrical breakdown of the insulation material. In subsequent tests to evaluate various insulator materials, the heaters burned in two, before power could be shut off. This not only prevented examination of the heaters to study the cause of the failure, but also resulted in damage to the instrumented test section.

An analog multiplier was set up to monitor the current and voltage applied to the high flux heaters and provide a continuous output signal proportional to electrical power for use with recording and control equipment. This permits direct time correlation between heater power and other test variables. A detector circuit was designed to monitor the power signal and cut off heater power if the electrical input power to the heater increases at a rate exceeding 2 kw/sec. A 1-kw power increase at a rate exceeding 2 kw/sec will cause heater power to be shut off

The power transient detector has operated satisfactorily and has shut off heater power in time to prevent destruction of heater cladding in each test in which it was used. This included more than 20 cases of electrical breakdown of heater insulation, which occurred during evaluation tests of various insulator materials. Tape recorded data indicated that the overall response time of the power transient detector system, including the relays which cut off heater power, was less than 1/20 sec.

The third stage in the development of the heater protection system included modifications to improve boiling detection sensitivity at high flow rates and permit testing of various length heaters without changing sensitivity limits.

Analysis of test data from the liquid superheat experiments indicated that the technique of using the signal from a permanent magnet EM flowmeter to detect boiling is completely satisfactory for purposes of heater protection. However, at high flow rates, the onset of boiling is so gradual and smooth that very fine boiling (as indicated by sound intensity from an accelerometer and wall pressure fluctuations from a recently installed Barton absolute pressure transducer with electronic output) sometimes occurs for several seconds before the signal from the flowmeter trips the comparator and cuts off heater power. As

described above, the liquid superheat tests are conducted by increasing power until the boiling detector cuts off heater power. Therefore, at the high flow conditions, the maximum power and maximum temperature readings which are recorded and used to compute liquid superheat, do not always correspond to the precise onset of boiling. Figure 5 shows a run at a flow velocity of about 2.5 ft/sec where heater power was shut off at the onset of boiling. The blip on the trace from one of the in-stream thermocouples is the point where heater power was shut off. Figure 6 shows another run at a flow rate of about 4.3 ft/sec. Note the more gradual increase in signal levels as boiling is approached. In an attempt to improve the accuracy of future liquid superheat measurements at high flow rates, two additional independent boiling detector circuits were added. The modified heater protection system includes the following:

- 1) The signal from the electronic pressure transducer is amplified, high pass filtered, and coupled to a comparator. Pressure transients observed at the outlet of the test section at the onset of boiling are damped sinusoids at about 6 cps. The NaK filled coupling lines, the bellows, and the mechanical linkage to the ferrite cam detector, apparently combine to act as a band pass filter which responds with high sensitivity to void formation. The comparator cuts off heater power if the oscillations on the observed pressure signal exceed a pre-selected threshold.

- 2) Signals from an accelerometer mounted on the test section are amplified and filtered to discriminate against background noise, and coupled to a loudspeaker in the test area. The filtered accelerometer signal is also coupled to an integrating circuit and comparator. The circuit is designed to cut off heater power if the integral of sound intensity exceeds a preset limit. Thus, spurious intermittent acoustic emissions associated with temperature changes should not trip the comparator; however, relatively steady acoustic emissions associated with boiling at high flow rates should trip the comparator within 2 to 3 seconds.

- 3) The signal from the EM flowmeter at the inlet of the test section is amplified, band pass filtered and coupled to a comparator with a variable delay. This portion of the electronic heater protection system was not modified.

- 4) The voltage signal from the ammeter shunt, in series with the high flux heater, is rectified, filtered, differentiated, and coupled to a comparator. This circuit, which replaced the power transient detector described above, is designed to detect sudden changes in the current applied

to the heater under test. The modified system automatically trips input power if heater current changes by more than 5% of full scale at a rate exceeding about 10% per second. Response time, including power relay, is less than 50 milliseconds. The change from a power sensing detector to a current sensing detector permits a single sensitivity setting to be used for any length graphite heater.

Figure 5. Accelerometer, Electromagnetic Flowmeter, Pressure Transducer, and Thermocouple Signals During Liquid Superheat Test at Flow Velocity of 2.5 ft/sec

Figure 6. Accelerometer, Electromagnetic Flowmeter, Pressure Transducer, and Thermocouple Signals During Liquid Superheat Test at Flow Velocity of 4.25 ft/sec

FUTURE APPLICATIONS

There are numerous references⁽⁹⁾ to the use of acoustic emissions^(10,11,12) from vibration pickups and neutron power oscillations from ion chambers⁽¹³⁾ to detect boiling, thermal mechanical instabilities, gas entrainment, and other abnormal conditions in reactor systems. However, the results reported in References 14 and 15 tend to indicate that future efforts in detecting many types of

abnormal conditions, should be made using ultrasonic techniques. The technique described in a previous section (passing accelerometer signals through filters and then to loudspeakers in the test area) was found to be a simple and effective method for helping operations personnel detect abnormal conditions during critical modes of operation. In Reference 15 it is indicated that this same technique becomes even more effective when selected ultrasonic sounds are translated (using a modulator technique) to the audio band and fed to the loudspeakers. This article asserts that many types of defects, leaks, etc. can be recognized by their ultrasonic sounds long before they worsen to the point where audible sounds are produced.

Experimental efforts currently underway at AI involve the use of accelerometers with resonant frequencies above 40 kc, and recording equipment with 200-kc bandwidth. Present plans call for use of this equipment to investigate resonant mechanical vibrations of internal components in an intermediate heat exchanger. Attempts will be made to detect periodic ultrasonic sounds of mechanical impacting.

The same equipment will also be used to study acoustic emissions of boiling, cavitation, and metallurgical flaws in test models; with the objective of developing simple, practical and reliable incipient malfunction detection systems for future test programs.

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Reactivity Noise Analysis As a Possible Tool for Diagnosis of Operational Malfunctions

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Extensive noise measurements have been carried out at Saxton, Carolinas-Virginia Tube Reactor, Yankee, Indian Point, and Trino Vercellese nuclear power plants. The spectral characteristics of power fluctuations of all of these reactors were similar in many respects, as has been reported in a previous paper,¹ however, some peculiarities were observed in the noise spectra in some of the plants. Later operational problems observed on these plants indicate that there might have been a relationship between the peculiarities in the spectra and the problems.

In noise analysis at CVTR, a sharp resonance was observed at 0.8 Hz, and several measurements were performed to investigate the effect of reactor operating conditions on the resonance. The resonance was found to be power dependent, moderator temperature dependent, and moderator flow dependent. The available data ruled out most of the possibilities considered except perhaps mechanical bowing. Several additional measurements were planned to obtain additional information at higher power levels and with high power density rods. However, these were not carried out due to premature failure of the special fuel rods. Postanalysis of the failed rods has not yet conclusively established the cause of failure.

Noise measurements at the Enrico Fermi plant at Trino Vercellese showed general falling off of the nuclear noise spectra at a slope of 40 db per octave above 1 cps similar to other reactors and two pronounced resonances between 10 and 15 Hz attributed to nucleate boiling. However, a resonance was also observed at 5 Hz more on one side of the vessel and less on the other. Based on later internal inspections, a possible relationship between the resonance and thermal shield problem was speculated.

At Yankee, Rowe a noise analysis program was carried out to study possible applications of these techniques for internal vibration. Direct study of the phenomena was not possible due to lack of appropriate instrumentation. Indirect measurements were attempted using available nuclear and flow instrumen-

tation. Measurements were carried out at different flow conditions (number of loops), power levels, and coolant temperature. No significant phenomena attributable to sustained oscillation of the active core relative to stationary structures were identified.

Noise measurements appear to have promise for the diagnosis of operational problems associated with power plants, especially if periodic measurements are carried out from the startup of a plant. No firm relationship has yet been established between the reported peculiarities in the noise spectra and the observed operational malfunctions.

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Neutron Density Fluctuations as a Reactor Diagnostic Tool*

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INTRODUCTION

The detection of anomalous reactor behavior by observing neutron density fluctuations was suggested several years ago.¹ However, to our knowledge there are still no operating power reactors with installed anomaly detectors that use this method. The lack of acceptance of the method is largely due to its being oversold at the beginning before any experience was gained. Neutron noise analysis will not necessarily detect every reactor anomaly, but it can be useful in detecting some types of reactor component malfunctions. The most attractive feature of noise analysis is that it is a nondisturbing measurement which does not perturb the normal reactor operation. The technique can best be described by comparing it with electrocardiograms used in the medical profession to detect heart defects. In general, a history of the patient (in our case the reactor) is kept so that when a defect shows up, the anomalous graph can be compared with normal graphs in order to diagnose the problem.

During the last year we have performed neutron density fluctuation analysis at two ORNL reactors, the High Flux Isotope Reactor (HFIR) and the Molten-Salt Reactor Experiment (MSRE). The most concentrated effort has been at the HFIR where more than 150 spectra representing 15 fuel cycles were obtained. These spectra were recorded to build a library of neutron fluctuation data that could be

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used in diagnosing reactivity anomalies in the HFIR and to aid in understanding the causes of reactivity fluctuations in reactors of this type. The basic objective at the MSRE is more specific in that we hope to be able to measure the relative amount of helium gas entrained in the fuel salt using neutron fluctuation analysis.

The overall objective of our noise analysis program is to develop an on-line reactor diagnosis tool that will not disturb the operation of the reactor but will still detect anomalous reactivity effects. Noise analysis has not been used extensively as an on-line diagnostic tool because of the inability to define the source of reactivity fluctuations in reactors. To better understand the source of these fluctuations, we have compared the predictions of detailed theoretical models with experimental data^{2,3} obtained under various reactor operating conditions. However, it still seems that the only way to apply noise analysis for reactor diagnosis is to obtain spectra either continuously or at frequent intervals to build a library of reference data that can be used for comparison when an anomaly occurs. This has been the main goal of our noise analysis program during the last year.

PROCEDURE USED IN NEUTRON FLUCTUATION ANALYSIS

The way in which we acquire and reduce our data has considerable significance in the interpretation of the results. Therefore a brief description of the data handling procedures will be given here.

Data Acquisition

Figure 1 shows the way that the time history of the neutron density fluctuations is recorded. The output of a neutron sensitive ionization chamber placed

Fig. 1. Neutron Density Fluctuation Analysis.

near the reactor is first amplified and then recorded on magnetic tape. An ac amplifier is used so that the dc signal from the chamber is rejected, since it contains no information of interest in fluctuation analysis. In general, the ac signal is only about 0.1% of the total signal and could not be amplified sufficiently unless the dc signal is first eliminated. The dc signal, measured with a current meter, is used as a normalization factor, as will be mentioned later. For our diagnosis measurements, a 15- to 30-minute time record is sufficient to yield results with statistical accuracy of approximately $\pm 10\%$.

Data Reduction

The magnetic tape is returned to the laboratory and analyzed as shown in Fig. 1. The tape is played back into a ten-channel analyzer⁴ which has ten plug-in filters tuned at different frequencies. The output of each filter is squared to obtain a signal that has a positive polarity only. Because the output of the squaring circuit is still a fluctuating signal (Fig. 1), the output signal is time averaged using a low-pass filtering circuit to obtain a slowly varying voltage that is proportional to the bandpass filter output signal. The outputs of the averaging circuits are sampled using a data logger, and the logged data (in the form of punched paper tape) are processed on a digital computer to correct for the frequency response of the data acquisition and analysis equipment. The digital computer output is a table of power spectral densities corresponding to the center frequencies of the bandpass filters used in the ten-channel analyzer. The frequency spectrum of the tape-recorded neutron fluctuation signal can be determined by replaying the tape through the analyzer several times and using different sets of filters for each analysis.

A typical spectrum as shown in Fig. 1 is made up of two parts. The part of interest for reactor diagnosis is called reactor correlated information. The correlated information (neutron fluctuations) is caused by small reactivity fluctuations in the reactor (for example, a vibrating fuel element or control rod). The flat portion of the spectrum contains no information for reactor diagnosis (it is due to the ionization chamber not being able to detect every neutron taking part in the chain reaction).

The frequency spectrum is normalized by dividing by the dc current squared. This normalization allows comparison of spectra taken under different conditions such as reactor power, ionization chamber sensitivity, or chamber position.

Data Interpretation

The most important step in reactor diagnosis, as in any experimental method, is a proper interpretation of the results. Our general approach is to compare spectra taken at different times and under different conditions. When spectra are compared, differences within the statistical error limits are ignored. In general we do not pay much attention to a change in the spectrum unless it is different by 50% or more. If a significant change in the spectrum is observed, an attempt is made to relate the result to some change in the operation of the reactor. Usually the differences seen are due to normal reactor behavior. However, in some instances, the abnormal spectra are due to real malfunctions of reactor components.

When neutron fluctuation analysis is used for reactor diagnosis, it is important to determine what is causing the reactivity fluctuations in a given reactor. This is not an easy task, since there are many mechanisms in a reactor that can cause reactivity fluctuations, such as coolant temperature fluctuations, pressure fluctuations, or mechanical vibration of fuel elements and control rods. To better understand the causes of reactivity fluctuations we develop detailed theoretical models of the reactors being studied. By comparing the shape and magnitude of the spectra calculated by use of these models with experimental spectra obtained at the reactors, we have been able to determine the main sources of reactivity fluctuations.

NEUTRON DENSITY FLUCTUATION MEASUREMENTS AT THE HFIR

Approximately 150 spectra were obtained at the HFIR during the 15 fuel cycles completed since the reactor first reached significant power level in May 1966. The HFIR was chosen for extensive study for two reasons. One, the HFIR is a high-performance system designed to operate at very high power density and therefore is interesting for noise analysis studies. The other more important reason for extensive studies at the HFIR is that histories on many fuel cycles can be obtained in a relatively short time, because a complete fuel cycle lasts only 22 days. This is extremely important in diagnosis work, because the success of the method depends on having a library of normal spectra and spectra for known anomalies before it can be useful for incipient failure detection.

Source of Neutron Fluctuations in the HFIR

After only two or three fuel cycles, we began to recognize a pattern in the HFIR spectra. Spectra taken during a normal fuel cycle are shown in Fig. 2. The figure illustrates how the frequency spectrum changes as fuel is burned and the control rods have to withdraw to maintain the reactor critical at 100 Mw. In general, this same behavior is seen in every fuel cycle. The decrease of the spectrum amplitude as shown in Fig. 2 is attributed to the decreasing reactivity worth of the rods as they are withdrawn. Therefore we concluded that most of the reactivity fluctuations are caused by flow-induced vibrations of the control rods.

To further substantiate our belief that the neutron fluctuations are caused primarily by rod vibrations, differential transformers were mounted on two of the five rods* and the signal was analyzed in the same way as neutron signals. These tests were performed with full coolant flow but with the reactor shut down. The transformers were mounted to measure the radial displacement vibrations of the rods. Figure 3 is a comparison of the mechanical rod vibration spectrum with a neutron fluctuation spectrum obtained during a normal fuel cycle at approximately the same rod withdrawal position. To compare the two spectra, they are normalized at 15 cps, and the neutron fluctuation spectrum is divided by the theoretical

*These tests were part of hydraulic studies performed by the HFIR Project.

Fig. 2. HFIR Neutron Fluctuation Spectra at Different Control Rod Positions during a Typical Fuel Cycle.

reactor gain squared to convert the neutron fluctuations to reactivity fluctuations. This division did not affect the observed shape significantly, since the neutron-to-reactivity gain of the HFIR is essentially constant[†] over the frequency range analyzed. Observation of Fig. 3 shows that the rod vibration spectrum has the same shape as the neutron spectrum. If the vibrations of all five rods were measured, the similarity might be more pronounced. Therefore, we conclude that the observed neutron density fluctuations in the HFIR are caused mainly by control rod vibrations and that we should be able to detect changes in the mechanical integrity of the control rods by observing the neutron fluctuation spectrum. The following discussion will illustrate how by using neutron fluctuation analysis techniques we were able to detect a control-rod bearing failure at the HFIR.

Fig. 3. Comparison of HFIR Control Rod Vibrations and Neutron Fluctuations Obtained at the Same Rod Position.

Anomaly Detections at the HFIR

We observed a departure from a normal spectrum at the end of September 1966 toward the end of the first 100 Mw fuel cycle (cycle 4). The output of an ionization chamber was recorded on magnetic tape on a number of days during cycle 4. At the end of the cycle, inspection of the control rods revealed that the upper guide bearings on one of the shim-safety plates had worn excessively. In addition, a number of the ball bearings had fallen out of the bearing race and were found in the primary inlet strainer. Analysis of the tapes at the end of cycle 4 showed a peak in the spectra that we had not observed before. This peak was observed in tapes taken 5 days prior to the normal reactor shutdown and discovery of the anomaly. It is worth noting that the reactor operating personnel observed no unusual behavior during these 5 days. Figure 4 shows spectra taken during fuel cycle 4, in which the bearings failed, and normal spectra obtained before and after cycle 4. The change in the spectrum due to the anomaly is most pronounced at 5 cps.

Fig. 4. Change in HFIR Neutron Fluctuation Spectrum as a Result of Rod Bearing Failure during Fuel Cycle 4.

These results, although encouraging, did not prove that fluctuation analysis could detect every failure of rod bearings. Most probably a different type of failure (or even the same type) would not be evident in the spectrum in the same way. However, during HFIR fuel cycle 15, the upper control rod bearings failed again. Analysis of the magnetic tapes obtained throughout this cycle shows a peak in the spectrum at 5 cps similar to the one observed during cycle 4. The anomalous spectrum is shown in Fig. 5, along with a normal spectrum obtained during a previous fuel cycle. This result leaves no doubt in our mind that neutron fluctuation analysis could have detected and in addition identified the upper rod bearing failure before operating personnel observed it with any of the existing reactor instrumentation. There is still no indication that other types of mechanical control-rod failures can be detected, but these results indicate that fluctuation analysis can be a useful tool for on-line reactor diagnosis.

Fig. 5. Change in HFIR Neutron Fluctuation Spectrum as a Result of Rod Bearing Failure during Fuel Cycle 15.

NEUTRON FLUCTUATION MEASUREMENTS AT THE MSRE

Neutron fluctuation analysis is being investigated at the MSRE as a possible means of measuring the relative amount of helium-gas voids in the circulating fuel salt. These voids are introduced into the fuel salt in the fuel-pump tank by the action of the xenon-stripping spray ring. Because the steady-state xenon poisoning in the MSRE is sensitive to the void fraction, it is impossible to properly interpret reactivity balance calculations⁵ unless the void fraction is known. Neutron fluctuation analysis may be useful for measuring the void fraction if the voids cause fluctuations in reactivity which in turn cause fluctuations in neutron density.

Preliminary measurements made in May 1967 indicate that the frequency spectrum of the neutron density fluctuations in the MSRE is sensitive to the void

fraction in the fuel salt. To perform these tests, the void fraction was purposely increased by lowering the salt level in the fuel pump tank. Neutron fluctuation data were taken when reactivity balance calculations showed a reactivity change due to voids. When the frequency spectrum of these data is compared with a spectrum obtained under normal conditions (Fig. 6), a difference is seen in the frequency region of 1 cps. This indicates that the neutron fluctuation spectrum might be sensitive to the relative amount of helium-gas voids in the fuel salt. However, before any definite conclusions were made, additional tests were run in which the cover-gas pressure on the system was changed. Figure 7 shows that

Fig. 6. Change in MSRE Neutron Fluctuation Spectrum when Voids were Introduced into Fuel Salt by Lowering Pump Bowl Level.

Fig. 7. Change in MSRE Neutron Fluctuation Spectrum when Helium Pressure in the Pump Bowl was Increased.

the peak at 1 cps decreases in amplitude when the helium pressure is increased. If the peak in the spectrum at 1 cps is due to helium voids, then its amplitude would not be as large if the pressure is increased. This is due to the voids being compressed, thereby decreasing the reactivity fluctuations. The results of this high-pressure test and the low pump-bowl-level test described previously indicate to us that the peak in the spectrum at 1 cps is due to voids. Therefore neutron fluctuation analysis should provide a method of measuring the relative void fraction in the MSRE fuel salt. But there is the possibility that other reactivity fluctuations in the MSRE are not caused by voids. In addition to these and other tests now being conducted, we are developing a detailed theoretical model of the MSRE which should enable us to predict the effects of voids in the fuel salt

and also to identify other sources of reactivity fluctuations in circulating fuel reactors of this type.

CONCLUSIONS

Because of our experience at the HFIR and MSRE, we can conclude that neutron fluctuation analysis can be a useful tool for reactor diagnosis. This technique is especially attractive as an on-line anomaly detection method because it does not disturb the reactor operation in any way. However, the usefulness of the method depends a great deal on having a reference library of normal spectra and abnormal spectra for known reactor anomalies in order to recognize similar anomalies.

As stated earlier, all of the spectra shown here were obtained off-line, but if fluctuation analysis is to be used as a diagnostic tool, it is imperative that the analysis be done on-line. Recent developments in digital frequency analysis using fast fourier transform algorithms* seem to hold the most promise for on-line analysis because of the flexibility in choice of analysis frequencies and speed of data reduction by on-line digital computers.

Also, in addition to neutron density fluctuations, noise analysis may be useful in analyzing other types of signals from a reactor system, such as pressure, temperature, mechanical vibrations, etc., that yield information about the behavior of the system. In some instances the fluctuations in the prompt gamma rays emitted in the fission process might contain the same information as the neutron density fluctuations.^{6,7} Comparison of a gamma spectrum (Fig. 8) with a neutron spectrum obtained at the same time indicates that the prompt-gamma spectrum has the same shape as the neutron spectrum, and therefore it should be applicable as a diagnosis tool. Analysis of the gamma spectrum could be very useful in systems where sensors must be placed far from the reactor vessel, resulting in attenuation of the neutron signals by shielding materials.

As stated previously, the basic problem in neutron noise analysis is the difficulty of identifying the origin of reactivity fluctuation in a reactor. Our experience has been that, although anomalous reactivity fluctuations can be detected by use of neutron noise analysis, the mechanism causing the unusual fluctuations is not always easily identified. Now that we are convinced that fluctuation analysis is useful for reactor diagnosis, we intend to employ the established technique of cross correlation to aid in determining the physical mechanisms that cause reactivity fluctuation. (Cross correlation determines that part of neutron noise caused by a particular variable, for instance, coolant temperature fluctuations.) We expect that these additional studies will help in identifying the mechanisms causing reactivity fluctuations in reactors and, therefore, further the usefulness of fluctuation analysis as a reactor diagnostic tool.

* J. B. Bullock paper, page 67.

Fig. 8. Comparison of Gamma and Neutron Power Spectra Densities Obtained Simultaneously at the HFIR.

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Discussion on Papers by Randall, Rajagopal, and Fry

ROUX (ORNL): Mr. Randall, have you found that mounting is a factor in the frequency response of your accelerometers? Saxe mentioned that in his acoustic work related to boiling detection the frequency response of his microphones was very sensitive to their mechanical mounting.

RANDALL (AI): Correct. The prominent frequencies that we detect are almost entirely those due to the resonant mechanical frequencies of the system which couples the acoustic emissions to the accelerometer. This includes the clamp with which the accelerometer is mounted. In most of these cases, about all I did was to mount the accelerometer on an angle bracket and clamp that angle bracket to parts of the system.

DUNEGAN (LRL): Mr. Randall, did you by any chance check frequencies near the principal resonant frequency of the accelerometer?

RANDALL: Yes. I found that by testing at the resonant frequency of the accelerometer, in an experiment with the intermediate heat exchanger, I was able to detect very low level mechanical impacting of internal components. By testing at the resonant frequency of the accelerometer, one obtains the highest possible efficiency because the accelerometer then looks like a narrow band-pass filter with a high gain at the resonant frequency. Also, in the ultrasonic region the signal-to-noise ratio tends to be better because there are fewer effects of pump disturbances and other things. The only noises in that frequency region are those related to cavitation effects, voids collapsing, and mechanical impacting of system components.

DUNEGAN: I was surprised that you did not operate in the ultrasonic region rather than at 3000 cycles/sec.

RANDALL: The amplitude of the signal near 3000 cycles/sec was approximately 10 times higher than in the ultrasonic region, although the signal-to-noise ratio was very good in the ultrasonic range. Furthermore, in the preliminary recordings it takes much less tape to record at the lower frequencies.

DUNEGAN: My second question is directed to all three gentlemen. I have heard today several papers related to the measurement of mechanical vibrations with neutron fluctuations. This seems to me to be an inefficient way of measuring mechanical vibrations, and I was wondering if you have thought of measuring the mechanical vibrations with standard vibration techniques?

RAJAGOPAL (WESTINGHOUSE): We realize that measuring neutrons is not the most direct way, but it is very difficult to install accelerometers in the hostile environment of a reactor core. On the contrary, the neutron signal is readily available.

DUNEGAN: My question was prompted by the control-rod mechanism failure mentioned by Fry. It seems to me that Fry must be able to attach some sensor to that control rod.

FRY (ORNL): We thought of attaching accelerometers to the rod drives which extend outside the reactor containment. Unfortunately, there is considerable mechanical damping between the control rod in the reactor and the rod drive on which we can mount the transducer. We felt that we might not be looking at the true picture if we did this. In some future cross-correlation measurements, we may try to cross correlate the output of a vibration sensor mounted on the control-rod drive with the neutron fluctuations.

GREEN (AEROJET): Mr. Randall, have you done any developmental work on sensors for use in a nuclear environment—both from a radiation standpoint and high temperature?

RANDALL: Personally, no, because I have been able to get the information I wanted at places that were not subject to high radiation levels. However, there is work being done at the Liquid Metal Engineering Center on strain gauges for in-core use at high radiation levels.

STERNLICHT (MTI): First, I want to comment that these noise-analysis techniques, while interesting, only seem to be valuable in one of two ways: Either one introduces malfunctions purposely and thereby experimentally obtains signatures that can be used for purpose of comparison, or one derives theoretical models on the belief that one will somehow be able to correlate experiments with theoretical analysis. Now, the latter approach is far better, because hopefully one will be able to extrapolate the results to other reactors.

Now, I would like to ask Mr. Fry these questions. What is the accuracy of your power spectra? Second, how good is the comparison between your theoretical analysis and experimental results? Third, how do you expect to apply your library of spectrum information to other reactors unless you establish some analysis and correlation between theory and experiment?

FRY: To answer the first question, our statistical precision is $\pm 10\%$ for the lower frequencies and improves as the frequencies increase since the filter bandwidths get larger.

In reference to the question of comparison of experiments with the theoretical predictions, we have been able to compare within a factor of 2 or 3. However, this is an oversimplified statement, and I would like to ask Jim Robinson to expand on this topic.

ROBINSON (UT): In this work one does not know in advance what the driving function is, and hence one cannot just plow ahead and perform one nice calculation unless one knows first what the driving function is. Even if the identity of the driving process were known, its frequency shape is probably not known unless measured. Hence our theoretical work has been a process of elimination. We first formulate a reasonably detailed system model and then postulate a list of possible driving functions with reasonable magnitudes. We then calculate theoretical power-spectral densities and compare these on a basis of amplitude and shape to measurements of absolute power-spectral density. Through this process we were able to eliminate 8 of 10 possible driving functions, because, to obtain theoretical results which matched experimental measurements, we would have had to invoke physically unrealistic driving functions.

STERNLICHT: In the theoretical analysis, not only does one have to know the driving function and its shape but also the attenuation characteristics, or the damping characteristics, of the system in order to predict absolute spectral densities. I would expect that nuclear systems which are composed of both fluids and solids would undoubtedly have damping characteristics and attenuation characteristics that one must also know.

ROBINSON: This is true, and we were quite concerned over this at one time because we were using a simple point-reactor-model-type analysis, which, of course, implies a damping factor of unity and therefore presupposes that it does not matter where the detector is put. To convince ourselves of the adequacy of the point model, we looked at different neutron detectors located in different positions, and, whenever we normalized

the spectra the way theory says it should be normalized, the magnitude of the spectra was independent of detector properties and detector location.

FRY: I would like to answer the last part of Mr. Sternlicht's question which concerned the extrapolation of these measurements to other reactor systems. I think that we have to treat each system as a new one. Because every system is individual and has a different set of driving and transfer functions, we cannot necessarily say on the basis of past success that we can detect a particular type of anomaly in a different system until we are able to accumulate a library of spectra on that new system.

DEBOISBLANC (IDAHO NUCLEAR): I think that we should not leave this topic with the impression that was created by the question of Mr. Dunegan: "Would it not be better to make direct mechanical vibration measurements?" The nub of the neutron-fluctuation approach is that the neutron signal contains the information from vibration, whatever or whenever the source; whereas, in order to detect vibration by the instrumentation of individual components, one must know in advance the physical location of the vibration source. After one has detected a vibration with noise analysis, then tracking down the source of the vibration is an entirely different process.

BALDERSTON (BOEING): I would like to point out that there are currently available a number of sonic transducers suitable for use in high-temperature ranges and radiation environments. We have employed such transducers at Boeing to detect such things as failure of ball bearings inside jet engines by mounting the transducer way out on the engine's external cowling. I cannot understand why you think there is a big problem in measuring a sonic vibration inside the reactor. Certainly there would be no more damping than we have with jet engines. Using sonic transducers on a hydraulic line, we have also detected problems back at the hydraulic pump.

RANDALL: I too have been amazed at the sensitivity of accelerometers which, when applied to any structural member that went into the core, were able to pick up what was going on way down in the reactor system. Coupling the sonic disturbances to accessible structures outside the reactor was no problem at all.

FRY: How do you get around the problem of all the confusing background noises that you have in this environment?

RANDALL: We make measurements of amplitude versus frequency versus time. We induce the different phenomena and find out what frequencies are most sensitive to the effect. We then pass the signals through band-pass filters and monitor their amplitudes or the amplitudes of their modulation as a function of time to detect a particular phenomenon.

FRY: But does not this approach imply that you know in advance every type of failure that you are going to experience?

RANDALL: No. The frequencies that are best for detection of these phenomena are generally related to the characteristics of the transmission medium that is involved. One generally gets the most detector response at those specific frequencies where there is optimum signal transmission from the source of the trouble — whatever it is — to the detector.

**Methods for Detection of Incipient
Failure in Reactor Primary Systems
by Nondestructive Testing**

B. H. Schofield and R. D. Wylie, Chairmen

Nondestructive Flaw Detection in Nuclear Power Installations

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Introduction

Acoustic signals are known to be generated in metals under stress. These signals emanate from so called "Healthy" or non-flawed materials as well as flawed materials. However, it has been observed that a certain characteristic "Burst" signal is indicative of material flaw growth. The threshold of appearance of the acoustic "Burst" signal, the increased occurrence rate of the signal with increasing material stress, and the acoustic energy release signature are all considered defining parameters of material flaw characteristics. Figure 1 represents the acoustic "Burst" signal as detected during an early test series.

Investigations are being conducted into the feasibility of applying the acoustic phenomena to flaw diagnostics. These investigations are directed at developing a non-destructive technique for the detection and definition of flaws in the large complex pressure vessels and primary system of nuclear power installations. They consist of applying the technique of detecting and interpreting acoustic emissions emanating from materials under applied stress.

Fig. 1 Half-Scale PM-2A Acoustical Burst Data.

Acoustic Test Program

The test program for application of the acoustic phenomena is directed at the detection and prediction of incipient failure in the primary systems of nuclear power installations. Phase one of the acoustic test program involved feasibility studies to determine if the acoustic signals generated by stress induced flaw growth in a complex nuclear pressure vessel, could be detected and utilized for flaw location. In support of these objectives, a simple stress wave detection, signal conditioning and recording system was assembled. This system was applied to the PM-2A reactor pressure vessel undergoing embrittlement studies at the National Reactor Testing Station.

The test program involved the hydrostatic stressing of the vessel under conditions of varying temperature and material flaw size. (Figure 2)

Fig. 2 PM-2A Pressure Vessel Test System.

The initial acoustic system detected some "Burst" type signals, but in general, the signals were obscured by background electrical and mechanical interference. During the early stages of the PM-2A program testing the acoustic system was developed to a stage where background noises were removed, resulting in a high acoustic signal to noise ratio. The system consisted of piezoelectric shear transducers and accelerometers (Figure 3), isolation amplifiers, filters to eliminate unwanted electrical and mechanical background noises, and a magnetic tape machine for data recording. Figure 4 represents a block diagram of the system. Figure 5 shows the signals derived from such a system.

The magnetic tape was post-test analyzed to relate the detected acoustic signal characteristics to applied vessel stress and stress rate. Figure 6 shows the acoustic count rate plotted versus vessel stress for the final test

Figure 3

Fig. 4 Typical Data Channel for PM-2A Acoustic Emission Recording.

Fig. 5 PM-2A Acoustical Burst Data - 3200 psi.

Fig. 6 PM-2A Acoustic Data for Test 8.

run of the PM-2A program. Note the peaking in the emission rate prior to vessel failure; this is interpreted as being the stress region where the growth rate is restricted by the build up of a large plastic zone at the leading edges of the flaw. The plateau observed from 2400 psi to 2900 psi is attributed to a change in the stress rate imposed on the vessel.

The conclusions of the phase I feasibility studies were as follows:

1. Acoustic signals could be detected emanating from the flaw region of a complex pressure vessel.
2. The "Burst" signals allowed location of the flaw through triangulation techniques. Figure 7 shows a time expansion of the acoustic "Burst" signal arriving at three separate transducers. This data provides time of arrival information for triangulation purposes.
3. The detection threshold for the "Burst" type signal, was 0.1 to 30% of normal vessel operating stress and 0.02% to 5% of the nominal material yield.

Fig. 7 PM-2A Acoustical Burst Triangulation Data.

In summary, it was concluded that with the proper detection, signal conditioning, and diagnostic system, acoustic signals could be detected, recorded, and analyzed to yield information regarding the presence of flaws in large complex reactor pressure vessels under low stress conditions.

In accordance with these conclusions, a second program phase was initiated. This phase was directed at development, fabrication, and test of an acoustics data acquisition system capable of detecting and recording all acoustic emissions emanating from the test specimen. That is, during the feasibility studies, two types of acoustic emissions had been characterized, the "Burst" signal indicative of flaw growth, and the low amplitude so-called "Continuous Emission" due primarily to stress induced dislocation movement. The upper signal trace in Figure 8 shows an acoustic

Fig. 8 Acoustic Burst and Continuous Emission Signals.

emission signal containing both the low and high amplitude signals. The lower trace is a time expansion of the signal region between "Burst" signals and shows the so-called "Continuous Signal."

In conjunction with the data acquisition system, a data analysis system, Figure 9, was constructed to display the acoustic data in a form that could

Fig. 9 Non-Digital Data Reduction Equipment.

be easily analyzed. Emphasis was placed on the development of the capability for interpreting the displayed data for flaw detection, location, and possible metal condition indications.

These tests were implemented with a series of tensile specimens and small pressure vessels, as shown in Figure 10. The test specimens were stressed under both flawed and non-flawed conditions. Additional investigations involved the stressing of small pressure vessels containing various

Fig. 10 Pressure Vessel Tensile Specimen.

degrees of annealing and embrittlement. The resulting data was analyzed for acoustic energy release and acoustic signature parameters that could indicate metal conditions. These tests indicated apparent differences in the total acoustic energy released from the annealed and embrittled vessels and gave some encouragement for use of acoustic data in metal condition interpretation. Figures 11 and 12 show the acoustic signatures

Fig. 11 Acoustic Data from Annealed SA-212 Pressure Vessel Test Specimen.

Fig. 12 Acoustic Data from Embrittled SA-212 Pressure Vessel Test Specimen.

derived from tests of identical vessels in the annealed and embrittled state.

Both detectors and instrumentation were modified during the test series to achieve a system with high sensitivity and a high signal to noise ratio. It was found that when the total acoustic signal was utilized, the signature described by the acoustic energy release rate followed that shown in the early PM-2A tests, but was statistically much richer in information, as can be seen in Figure 13, All acoustic signatures followed, in general,

Fig. 13 Acoustic Data—Carbon Steel Tensile Specimen.

the dislocation mobility theory for materials under applied stress. That is, the maximum density of mobile dislocations reaches a peak at a relatively low strain level and decays exponentially with increasing strain.

The application of the acoustic analysis technique to operating nuclear power installations represents a variety of problems imposed by such environmental conditions as unpredicted mechanical and electrical noise, radiation, heat, and the usual inaccessibility of a typical reactor primary loop.

Test phase III was thus directed at a series of tests to determine the feasibility of applying the acoustic phenomena for integrity analysis of the primary system of an operative nuclear power installation.

Noise analysis investigations were conducted on various NRTS reactor systems and the acoustic system was modified to eliminate or limit the unwanted noise which complicated the acquisition of desired acoustic data.

With the results of the above described program, an in-plant acoustic analysis system, shown in Figure 14, was fabricated by April 1967. The first

Fig. 14 Signal Detection and Condition System.

opportunity to test this system on an operative nuclear power installation was presented by the SM-1A pressure vessel anneal program conducted in the summer of 1967 at Fort Greely, Alaska.

The objectives of applying the acoustic system to the SM-1A were essentially threefold. First, to evaluate the feasibility of using the acoustic phenomena under the environmental conditions imposed by an operative nuclear power installation; second, to evaluate in-plant system installation techniques, and third, to evaluate the feasibility of

determining the change in embrittlement of the primary pressure vessel utilizing the recorded acoustic data.

The acoustic data acquisition on the SM-1A was accumulated in the hydrostatic test mode due to the anticipation of noise interference generated by turbulent fluid flow in the primary system.

A total of twelve transducers were epoxy bonded to the primary system located within the vapor container. The transducers were located outside the primary vessel shielding envelope on the primary system inlet and outlet duct (approximately 8 feet from the reactor vessel). They were also located on the control rod drives extending approximately 10 feet below the reactor vessel, and on the top head of the vessel under 13 feet of shielding water. The transducers were mounted by use of BR 600 epoxy; the exception was the top head transducer which was magnetically mounted. Figure 15 shows the general location of the transducers on the SM-1A primary system. Figure 16 is a photo of the transducers on the reactor outlet duct.

Fig. 15 SM-1A Power Plant.

Figure 16

The transducer environmental conditions were defined as 100 Rem/hour of radiation, and 573°F of temperature. During the test, the unexpected condition of a saturated atmosphere was induced by a leaking steam valve within the vapor container.

The transducer signals were transmitted, by the use of preamplifiers through 100 feet of coaxial cable, and a vapor container penetration, to the signal conditioning and recording systems located on the ground floor of the power plant. Figure 17 is a photo of the signal conditioning and data reduction systems.

Figure 17

The total acoustic signal was recorded on magnetic tape while simultaneously being plotted by an on-line analysis system. Figure 18 shows the data reduction system in block diagram form, but does not show the X.Y.Y' plotting systems used for on-line analysis of the total acoustic signal. The "Burst" signals were processed by a digital analysis system and printed out in terms of "Bursts" per unit time, versus transducer channel and vessel stress.

Fig. 18 Data Acquisition System.

Several problems were involved in the SM-1A tests that had not previously been encountered in the PM-2A and laboratory vessel tests. The 1st problem was induced by the slow stress rate involved in the hydrostatic test. Stressing of the vessel was achieved through use of the plant pressurizer, supplemented by core heat; the maximum stress rate attainable was 0 to 1200 psi in 4 1/2 hours. To overcome the low acoustic energy release rate produced by the vessel stress rate, high system sensitivity was necessary; accordingly, signal amplification was set at approximately 10^6 .

The 2nd problem was that of locating the acoustic "Burst" signal sources through triangulation techniques utilizing detectors located several feet from the vessel on attached piping. To simplify the problem, a calibration procedure was developed using simulated acoustic pulses induced at known locales on the vessel appendages. The signal arrival time from these simulated acoustic "Bursts" were recorded on all transducer channels. Knowing the velocity of signal transmission, all transducers could be physically located relative to each other. The calibration also provided a means of determining transducer sensitivity.

The acoustic test program involved the accumulation of acoustic test data in three stages. First, accumulation of data during a hydrostatic test prior to pressure vessel anneal. Second, accumulation of data during the anneal operation, and third, accumulation of acoustic data during a hydrostatic test following the pressure vessel anneal.

The first two phases of the test program were accomplished; however, the third phase has been deferred until December 1967, due to plant operational problems.

The data accumulated during the pre-anneal hydrostatic test was beyond expectations and included clean, clear signals of both the low amplitude "Continuous Emission" and the high amplitude "Burst" signals.

Preliminary analysis of the "Burst" data has been achieved by periodic sampling of the 4 1/2 hour pre-anneal test data. By using time expansion techniques, time of arrival data has been derived from the sampled data. Figure 19 shows a time expansion of an acoustic "Burst" signal occurring at six separate transducer locations. Manual triangulation indicated 65% of the signals to be coming from two sources in the upper half of the pressure vessel, as shown in Figure 20. Ten percent of the remaining signals were defined as coming from several separate locales in the bottom half of the vessel. The remaining "Burst" data, approximately 25% were from source locations external to the vessel, and were tentatively being attributed to turbulent fluid noise generated by various pumps operated in conjunction with the primary system.

The total acoustic emission signal as derived from on-line analysis and confirmed by post analysis of the magnetic tapes, showed a direct relationship of acoustic energy release rate, Figure 21, with vessel stress rate, Figure 22. As determined in earlier laboratory work, the total acoustic energy released is expected to be a function of the total stress and not the function of the stress rate.

Figure 19

Fig. 20 SM-1A Acoustic Source Location

Fig. 21 SM-1A Pre-Anneal Acoustic Test Data.

Fig. 22 SM-1A Pressure Vessel Hydrostatic Test, Stress vs Time.

It is anticipated that the "Continuous Emission" energy release signature of the pre-anneal hydrostatic test will yield information on the degree of change in vessel embrittlement, when compared to the post-anneal acoustic signature.

The "Burst" data derived from the digital analyzer indicated the same trend as shown by the total acoustic signal; however, it is statistically poor.

Data accumulated during the anneal operation, which required primary system fluid flow, indicated that large "Burst" signals could be detected above the turbulent fluid flow noise. However, the data is statistically poor and its use, for even large source locations, is questionable.

The data accumulation and subsequent preliminary analysis associated with the SM-1A test program have thus far indicated that the present acoustic data acquisition system and associated analysis techniques, used in the hydrostatic test mode, allow detection of both the "Continuous" and "Burst" acoustic signals under the environmental conditions imposed by an operative nuclear power installation. With this data, it has been shown that source location can be achieved on a complex system with transducers located remotely. The feasibility analysis using acoustic signature and energy release valves for determination of metal condition has been delayed until completion of the test program; that is, data accumulation and interpretation during a post-anneal hydrostatic test.

The results of the in-plant test phase have indicated the need for an automatic data reduction system for flaw location. That is, approximately one hour of effort is required for the data reduction and triangulation analysis associated with one acoustic "Burst" signal. In accordance with this requirement, an automatic source location system is being fabricated at this time and will be available for data reduction within three months.

Extensive analysis of accumulated acoustic data has indicated that information regarding flaw size and growth rate may be derived by application of acoustic emission behavior models presently being investigated.

Conclusions

Investigations and feasibility studies involving the application of the acoustic analysis phenomena for flaw detection, flaw definition, and incipient failure prediction have led to the utilization of an operable in-plant analysis system. This system has shown capability for detecting and locating flaws in the complex reactor pressure vessel and primary loop piping of operative nuclear power installations. Due to fluid flow noise interference, it is limited at this time to application in the hydrostatic test mode; however, the significance of this system should not be disregarded. Even in its present stage, acoustic analysis provides a valuable tool for inspection of pressure vessels and pressure piping. Acoustic analysis techniques coupled with the search and find techniques of ultrasonics and radiography will provide a valuable tool in the quality control of vessel manufacture and as an in-plant integrity assurance safeguards tool.

The long term goal of developing an incipient failure diagnostics system for nuclear power installations appears feasible, but will require a program of thorough test evaluation, and development prior to its achievement.

Stress-Wave Analysis Technique for Detection of Incipient Failure

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ABSTRACT

Research has shown that crack growth resulting in a sudden release of elastic energy can be detected by a sensor that converts the elastic wave into an equivalent electrical signal. Thus, this technique is unique as a nondestructive inspection method in that the growing defect in the material transmits its own signal, with the sensor acting as the receiver. The receiving sensors used in our studies and applications have been piezoelectric accelerometers, directly calibrated in engineering units.

The Stress-Wave Analysis Technique (SWAT) has been employed as a research tool aimed at learning more about the causes of slow crack growth, particularly as affected by fatigue, hydrogen embrittlement, stress corrosion cracking and spontaneous strain aging. SWAT is now being employed as a field inspection and monitoring process during the pressure qualification testing of rocket motor cases and other pressure vessels. With SWAT one can not only determine if there is a growing flaw, but can also locate the flaw by triangulation techniques. Moreover, in most structures as the crack increases in length, there is an increase in the rate and amplitude of stress-wave emission that presages failure.

I. INTRODUCTION

In recent months great interest has arisen in the possibility of applying stress-wave-emission (SWE) techniques as a nondestructive inspection method for assuring the structural reliability of system

components. This paper will trace the background of programs and the development of the current capabilities of the stress-wave analysis technique (SWAT) and the projected development for new applications and research programs. In addition, extensions of the technique to other objectives will be mentioned and finally, an extensive bibliography for those interested in a more complete background in the state-of-the-art is included.

II. BACKGROUND

The two overall objectives of all investigations and applications of the stress-wave analysis technique, accomplished by Aerojet, have been the prevention of catastrophic failure and the increased reliability of the structure in use. Our primary applications effort has been devoted to the detection and location of growing flaws by stress-wave emissions. In research the primary objective has been to learn more about the causes of slow crack growth, particularly as affected by fatigue, hydrogen embrittlement, stress corrosion cracking, and spontaneous strain aging. Few, if any, structures are totally free from defects which propagate in service. Through stress-wave emissions, the incremental extensions of a flaw can be detected and located by triangulation.

Stress-wave emissions are caused by crack growth and fracture processes such as deformation and dislocation movement. These waves can be detected by any of a number of different types of sensors. At Aerojet standard piezoelectric accelerometers are used as the stress-wave sensors. These mass-loaded crystals deform due to the movement of the specimen surface and the inertia of their mass, thus producing an output signal. These sensors have characteristics of flat frequency response over their rated range and output signals directly related to the acceleration units of g's. Other investigators have been using nonmass-loaded sensors to obtain higher frequency response.

Minute failure processes⁽¹⁾ can be detected as a continuous type of emission with very sensitive instruments (sensors), high amplification, high frequency response, and a relatively noise-free environment. Characteristics of this type of emission have been correlated with such engineering values as the tensile yield strength of the test specimen.⁽²⁾ Continuous emissions are generally considered to be created by deformation and associated dislocation pile up occurring when the metal is strained.

The type of stress-wave emissions which are being used for structural mechanics, materials research, and product quality assurance and reliability are the larger, sporadic burst type of emissions directly attributable to flaw extensions of a macro-size. These are flaw extensions which are capable of being analyzed in terms of nondestructive inspection, structural design and analysis, and materials research.⁽³⁾ It is this type of stress-wave emission that is discussed in this paper.

A. APPLIED TECHNOLOGY

1. Pressure Vessel Burst Strength

The first application of stress-wave emission at Aerojet dealt with the determination of the remaining structural integrity of a rocket motor case after its manufacturing proof test was

completed. This program was conducted to determine whether a relationship existed between the energy released during the proof pressurization test of glass-filament-wound rocket motor cases and the structural integrity of the case after the test.⁽⁴⁾

A stress-wave detection and analysis technique was developed to predict the burst strength on the second cycle of pressurization based on the "stress-wave-signature" obtained from the data acquired during the initial hydrotest cycle. This relationship between the noises emitted and the structural integrity of the case has been explained in terms of a statistical "failures per unit area" concept. The concept is based on the premise that equal cross-sectional areas of the same material can withstand an equal number of discrete failures (all other factors being held constant). The concept is applied by measuring the energy released during a series of replicate specimen tests and relating this value to the strength of the specimen. A series of such tests provides a statistical basis for establishing the relationship, which is then projected, by simple ratio, to other test specimens of the same material and configuration.

2. Triangulation

The program previously described inevitably led to the triangulation of SWE. Since the energy released during incremental failure processes is detected as a stress wave which travels through the structure at a velocity characteristic of the particular material, sensors attached to the test specimen receive the wave front at different times depending on the distance from the flaw to the sensor. These time differentials are used to triangulate to the origin of the waves, thus locating the growing flaw. The analogy between this application of the stress-wave technique and earthquake seismological techniques is evident. In Reference 5 graphical and analytical methods for triangulation on cylindrical pressure vessels are given.

It should be clear to the reader, at this point, that the ability to triangulate to the origin of the stress wave depends only upon the wave being received by a suitable number of sensors (a minimum of two if in an array such as on a linear weld seam). The significance of the wave itself, in terms of the criticality of flaw, is an independent consideration. The majority of these triangulation applications have been on high-strength metal pressure vessels and glass-filament-wound rocket motor cases.

3. Locating the Source of Pressure Vessel Failures

Through the use tape recorded data and posttest triangulation methods, the origin of stress-wave emissions have been located in a large number of pressure vessels. Table I provides a short tabulation of pressure vessels which were failed in testing and the distance from the stress-wave-defined origin to the metallurgically defined origin of failure.

During this period of time SWAT was being applied to various pressure vessel tests in order to determine the in-field feasibility of the system. Tape recordings were made during the manufacturing proof test of several large rocket motor cases in order to obtain data regarding wave propagation, velocity, attenuation, and

Table I SWAT Location of Failures (Metal Pressure Vessels Only)

Chamber	Material	Diameter, in.	Length, in.	Number of Stress-Wave Sensors	Distance from Stress-Wave Origin to Metallurgy Defined Origin of Failure, in.
Large Solid Rocket	18% Nickel, 250 grade maraged steel	260	690	24	12
Minuteman	6Al-4V Titanium	52	110	5	2.5
Large Solid Rocket (Subscale)	18% Nickel, 200 grade maraged steel	36	100	7	2.0
Experimental	AMS-355 Steel	26	156	10	0
Experimental	AMS-355 Steel	26	156	10	0.5
Minuteman	6Al-4V Titanium	52	110	5	2.5
Large Solid Rocket (Subscale)	18% Nickel, 200 grade maraged steel	36	100	7	1.75
Service Propulsion System	6Al-4V Titanium	45	160	12	2.5

other experimental data. It was during one such data gathering application of SWAT that a large pressure vessel failed. This incident was unfortunate in that the SWE state-of-the-art did not include an on-line system at this time. Responsive action to stress-wave indications during the test was neither programmed nor required during this test.

The fact that the chamber did fail at approximately 50% of its scheduled pressure level due to undetected flaws merely points up a reason for the advancement of stress-wave emission technology. The failure report ⁽⁶⁾ notes that many defect areas were found that had not been evident prior to the test.

The stress-wave data recorded during this test showed at least two large amplitude stress waves had originated from the failure area. Although the nearest stress-wave sensors were a considerable distance from the flaw which originated failure, the triangulation of the failure origin was accomplished prior to the metallurgical investigation. Figure 1 shows the size of flaws which caused this failure and also shows the SWE triangulation solution for the origin of failure.

Other pressure vessels, which had been rejected by their manufacturer as unsalvageable due to gross porosity in three weld seams, were tested to failure in low-cycle high-stress fatigue. SWE data recorded during each pressure cycle were used to locate the areas of maximum flaw growth. Near the end of the test program, radiographs taken of the weld seams between each pressurization cycle, which had shown little or no evidence of flaw growth in the early cycles, began

Fig. 1 Fracture Path at Origin of Failure.

to show the interaction between pores. Both chambers failed at pressures near the design strength thus showing no short-term structural degradation due to the gross porosity. The stress-wave emission data was used to confirm the location of primary failure for these pressure vessels.

4. Stress-Wave Emission as a Signature

The feasibility of a system which used the elastic waves associated with flaw growth to indicate potential failure and initiate remedial action to prevent these failures was demonstrated on a program for NASA.⁽⁷⁾ In this program a series of tensile specimens were tested to failure while recording SWE. It was during this program that the characteristic increase in rate of SWE with approaching failure was noted in metallics. This stress-wave signature was then used to abort the constantly rising uniaxial load, on a series of tensile specimen tests, and prevent the specimen from completely failing. The fact that flaw growth proceeds incrementally and can be arrested through prompt responsive action was thus confirmed.

This rate-of-SWE signature had been noted as a presage of failure in the glass-filament-wound pressure vessel tests, but never before had it been used as a criterion for aborting and maintaining the integrity of the structure. The data shown in Figures 2 and 3 are stress-wave emissions versus time (i.e., pressure, load) for a glass-filament-wound pressure vessel and for a 200 grade maraging steel tensile specimen. The increased rate of SWE can be observed in both figures.

Similarly, additional programs have also demonstrated this SWE characteristic. The stress-wave data shown in Figure 4 are

Fig. 2 Stress-Wave Emission Record of a Hydrostatic Test (Glass-Filament-Wound Pressure Vessel).

Fig. 3 Stress-Wave Emissions vs Time (200 Grade, 18% Nickel Maraging Steel, Tensile Specimen).

CONTINUOUS
DATA RECORD
1000 LB TO
39,000 LB (FAILURE)

TIME BASE
EXPANSION DATA
RECORD OF
LETTERED AREAS
ABOVE.

Fig. 4 Stress-Wave Data vs Specimen Load (0.375-in. Aluminum Alloy Material).

SWE versus time for a part-through-crack surface flaw tensile specimen tested under constantly increasing load directly to failure. The figure was photographed directly from an oscillographic recording of the original tape recorded data for a tensile specimen of 2014-T6 aluminum alloy.⁽⁸⁾ The increased rate of SWE is distinctly noted.

The similarity of the failure process is noted even in less common engineering materials such as glass. The oscillographic data shown in Figure 5 is a reproduction of the SWE from a hollow glass hemisphere undergoing hydrostatic pressure failure tests.⁽⁹⁾ This chemically strengthened glass specimen also shows the characteristic increased rate of SWE prior to failure. This increase is displayed in Figure 6 for three glass hemisphere specimens. The stress-wave signature was used to unload test specimen 3; structural damage was clearly observable in the form of chipping, spalling, and cracking.

5. Measurement of the Failure Process

The inability of radiographs to determine the amount of flaw growth associated with stress-wave emissions required that we develop other techniques to resolve this question. A cross section through a PTC tensile specimen, one of several methods used to correlate SWE and flaw growth, is shown in Figure 7. The constantly increasing uniaxial-applied load on this specimen was released after large numbers of stress waves were observed and the specimen was then cycled in high-frequency fatigue at a much lower stress level than previously applied. This fatigue cycling was done in order to extend the existing crack boundary and thus provide a fatigue surface marking which clearly defines the amount of slow crack growth that had occurred during the uniaxial tension test and created that particular amount of stress-wave activity. This procedure was repeated and then the specimen was taken to complete failure.

In still other tests unflawed samples were loaded and then unloaded prior to stress-wave emission, and also after one or several waves were observed. Each specimen was then sectioned and metallurgically examined in order to define the relationship between stress-wave emissions and flaw extensions.⁽¹⁰⁾ Through techniques such as these, SWE flaw-growth relationships have been determined.

6. Effects of Environmental Exposure

Tests have been conducted on pressure vessels subjected to corrosive environments. A titanium pressure vessel, whose mating tank had failed in service after being exposed to a corrosive fluid, was pressure tested with SWAT applied. Prior to test the tank had been inspected with a fluorescent penetrant but no indications of surface defects were found.

The stress-wave data recorded prior to vessel failure coincided with the failure origin. Posttest inspection of selected areas in the vessel were performed using high-resolution penetrant inspection materials. The only indications of surface defects were found throughout the aft hemisphere containing the fracture origin (as determined metallurgically and by SWE triangulation).⁽¹¹⁾

Fig. 5 Stress Waves vs Pressure (10-in.-dia Hollow Glass Hemisphere).

Fig. 6 Stress-Wave Emission Rate vs Hydrostatic Pressure (Three Test Specimens).

Legend:

a - Electrical Discharge Machining Notch	e - Single cycle tensile crack extension
b - Bending fatigue precrack	f - Bending fatigue crack extension
c - Single cycle tensile crack extension	g - Plane-strain fracture area
d - Bending fatigue crack extension	h - Shear lips

Fig. 7 Fracture Surface of Part-Through-Crack in Aluminum Alloy.

B. APPLIED RESEARCH

Realization of the full potential of the Stress-Wave Analysis Technique for the detection of incipient failure can only be accomplished if the true relationship between the SWE and permanent deformational processes is known. Aerojet has devoted a considerable portion of its SWE effort into defining the relationship between SWE and the criticality of the flaw. Emissions coming from microcrack origins, which may never develop into engineering size flaws compromising the integrity of the structure, must be separated from those emission of growing defects which are hazardous.

1. Causes of Subcritical Crack Growth

a. Instrumentation and Test Method

In most of the metallurgical research, a fatigue-precracked, single-edge-notch (SEN) tensile specimen is used (Figure 8). The edge notch is specially machined to accommodate a double-cantilever crack-opening displacement gage. When calibrated, this device permits one to determine the overall length of the growing crack at any given instant. The output of this gage is continuously displayed on a strip chart recorder. With the strip chart moving at a constant speed, it is possible to determine from the output of the gage the crack growth rate over any given period of time. A second method for monitoring crack growth is by stress-wave emissions.

Fig. 8 Single-Edge-Notch Tensile Specimen Instrumentation.

During tests the signal from the accelerometer is usually recorded on magnetic tape as raw, unfiltered data. Thus on subsequent playback of the tape recording, filtering, time-base-expansion, spectrographic, and other data analysis techniques may be accomplished. The tape recorder allows sinusoidal signals as high as 300 KHz to be faithfully reproduced.

In studies of crack growth the number of signals per unit time or the time between signals provides a measure of crack incubation time. Where time-dependent metallurgical phenomena such as hydrogen embrittlement, stress corrosion cracking, and spontaneous strain aging are involved in the crack-growth process, the measurement of crack-incubation time appears to be an important key to a better understanding of crack-growth mechanisms.

By monitoring the wave motion resulting from the sudden release of elastic energy associated with discontinuous crack growth, it is possible to detect each individual increment of crack growth. Conventional methods like the crack-opening displacement gage are useful for detecting macroscopic movements of the crack; whereas the stress-wave analysis technique permits the measurement of crack movements as small as 10^{-5} sq in. at the sensitivity levels used in the Aerojet system.

b. Detection of Microcracks by Stress-Wave Emission

One of the metallurgical researches was conducted for the Air Force⁽¹⁰⁾ to investigate the practical applicability of the stress-wave emission technique to the detection of crack initiation and growth in high-strength structures. Tests correlating stress-wave emission with deformation and fracture processes were conducted on vacuum-arc-remelted grade-200 and grade-300, 18% nickel maraging steel, and D6aC low-alloy steel. The D6aC was heat treated to provide a high-strength low-fracture toughness condition (600°F temper) and a low-strength high-fracture toughness condition (1075°F temper). Presumably these two conditions would span the normal conditions of usage of the material. The 18% nickel maraging steels were heat treated using the normally recommended aging treatments (900 to 920°F for 3 hr).

The results obtained showed that stress-wave emission monitoring techniques could be employed to detect crack initiation and growth in the materials evaluated. The stress-wave emission parameters most useful for this purpose were stress-wave amplitude and rate of occurrence. Frequency spectrum and pulse shape were shown to be characteristic of the type of accelerometer and mounting procedure and, consequently, gave relatively little information concerning the fracture process.

It was further shown for both classes of material that gross plastic deformation, fracture of inclusions, small microcracks corresponding to a grain dimension in length, and the small amount of crack growth accompanying the formation of the plastic zone at the tip of a fatigue precrack do not result in stress-wave emissions at the 0.01-g level of sensitivity. Figures 9 and 10 illustrate the types of cracking under investigation.

In general, in the materials evaluated, the formation of a crack with area greater than 10^{-5} sq in. resulted in the

Fig. 9 Fractured Inclusions and Cracks on the Tensile Surface of Maraging Steel Specimens.

emission of stress waves; whereas, formation of a crack with an area less than 10^{-5} sq in. did not result in detectable stress-wave emissions at the 0.01-g sensitivity level. The material type and heat treatment appeared to affect the type of microcracking which occurred, but with no major effect on the size of crack necessary for detectable stress-wave emission. Over the range of properties tested, higher-strength, lower-fracture-toughness material conditions were accompanied by a lower number of high amplitude emissions in comparison to lower-strength, higher-toughness material conditions.

The effects of varying test temperature from -100 to $+250^{\circ}\text{F}$ and varying strain rate from $\epsilon = 0.01$ to $\epsilon = 0.0001 \text{ min}^{-1}$ were investigated. The stress-wave emission decreased below room temperature for the D6aC material, but was relatively unaffected by temperature in the other materials. The D6aC- 600°F temper specimens tested at -100°F (apparently below the ductile-brittle transition) fractured at

Crack Too Small To Give Detected Stress Wave

(a) Initially Polished Surface (160 X).

(b) Underlying Crack.

Cracks Giving Stress Waves

(c) Initially Polished Surface (160 X).

(d) Section Parallel to Tension Surface (160 X).

(e) Section Perpendicular to the Tension Surface (375 X).

Fig. 10 Surface Markings and Underlying Crack on Tensile Surface of D6aC (6000F Temper Specimen).

the first stress wave. Multiple stress-wave emissions preceded failure of the D6aC 600°F temper conditions at all test temperatures from -100 to +250°F. The total number of stress-wave emissions appeared to be primarily dependent on the total amount of strain and not significantly on strain rate within the rates investigated. However, the rate of stress-wave emission was dependent on the strain rate, an order-of-magnitude decrease in strain rate being accompanied by an order-of-magnitude decrease in the rate of stress-wave emission.

The effect of surface finish on stress-wave emission was not significant when considered in terms of the variation in stress-wave emission amplitude and rate of occurrence between duplicate samples. Conversely, surface oxides tend to fracture under stress giving stress-wave emissions that may tend to give false indications of flaw growth. In practice, however, these emissions would probably be random with respect to location, and because of this characteristic, they could be separated from the multiple stress-wave emission associated with the growth of a dangerous flaw in a structural component.

c. Stress-Wave Characteristics at Crack Instability

Another metallurgical research was conducted for the Office of Naval Research⁽¹²⁾ to determine (1) if there is some recognizable, reproducible stress-wave characteristic (or characteristics) which predict the onset of crack instability under plane-strain and plane-stress conditions; (2) if such characteristic varies from one material to another; and (3) if such characteristic is affected by specimen configuration.

From single-edge-notch (SEN) tensile tests of 7075-T6 aluminum, HY-80 and HY-150 constructional steels, D6aC low-alloy high-strength steel, and 6Al-4V titanium, it was found that there are recognizable, reproducible stress-wave characteristics at crack instability. The characteristics are: (1) for plane strain instability (pop-in), a marked increase in stress-wave amplitude and (2) for plane-stress instability, a marked increase in both the stress-wave amplitude and rate of occurrence. Tests of various specimen widths, thicknesses, and notch depths in 7075-T6 aluminum indicated that the above characteristics persisted in spite of changes in specimen size and configuration (Figures 11 and 12).

In highly tough D6aC steel (austenitized at 1750°F and tempered at 1100°F) and low-interstitial 6Al-4V titanium (solution treated at 1750°F and aged at 1250°F), there was no detectable pop-in on the COD chart. There was, however, a deviation from linearity in the COD chart which is normally interpreted as the development of a large plastic zone and/or crack growth. Although there were no large stress waves in the load range corresponding the deviation from linearity, there was an increase in the stress-wave count at approximately this load. Furthermore, with increasing deviation from linearity in the COD chart there was an increasing stress-wave count continuing up to fracture (Figures 13 and 14).

In the highly tough HY-80 and HY-150 constructional steels, again there was no detectable pop-in either by COD or SWE. Fracture occurred by microvoid coalescence (Figure 15) in such small increments that at the lower stress levels used in testing HY-80, two orders of magnitude higher sensitivity had to be used. Both the HY-80

Fig. 11 Stress-Wave Count vs Specimen Thickness (7075-T6 Aluminum).

Fig. 12 Stress-Wave Count Rate vs Notch Depth (7075-T6 Aluminum).

Fig. 13 Stress-Wave Count Rate vs Heat Treatment (D6aC Low-Alloy High-Strength Steel).

Fig. 14 Stress-Wave Count Rate vs Aging Temperature (Ti 6Al-4V).

6800 X

5000 X

Fig. 15 Fracture by Microvoid Coalescence (Electron Fractographs of Slow Tearing in Specimen 80-2 during 40-Kip Hold).

and HY-150 showed a marked increase in the stress-wave count as final instability was approached. Furthermore, during periods of sustained load, the stress-wave count rate in both the HY-80 and HY-150 decreased exponentially with time at load, indicating a microcreep process (Figure 16). In the low-interstitial, 0.125-in.-thick, 6Al-4V titanium

Fig. 16 Stress Wave Emission and Crack Growth vs Elapsed Time (HY-150 Steel during Sustained Load at 55 Kips).

tested, there was neither COD nor SWE evidence of plane-strain pop-in. The final instability, however, was clearly marked by an increase in SWE amplitude and rate. Aged at 1250°F to produce maximum toughness, 6Al-4V was approximately at the same strength as HY-150 steel. Thus, in the thicknesses tested, these highly tough constructional alloys did not undergo plane-strain fracture; however, as the final instability was approached, both stress-wave amplitude and rate of occurrence served as precursors of failure (Figure 17).

In the more brittle D6aC high-strength steel (austenitized at 1750°F and tempered at 600°F) and in 7075-T6 aluminum, plane-strain pop-in was identified by an order-of-magnitude increase in stress-wave amplitude and the plane-stress instability was identified by an increasing stress-wave count rate, generally starting at approximately the plane-strain pop-in.

In the extreme embrittlement of D6aC when inadequately austenitized at 1550°F, fracture occurred directly from plane-strain pop-in with only two or three stress waves occurring before

Fig. 17 Stress-Wave Emission vs Increasing Load
(Rate and Amplitude in HY-150).

the final unstable burst of crack growth. With prior knowledge of such low fracture toughness, one would hesitate to allow more than one or two stress waves before unloading in service.

From this research it was concluded that crack growth can be detected and converted into an electrical signal, and that such signals have recognizable, reproducible characteristics which serve to identify plane-strain instability and presage plane-stress instability.

d. Metallurgical Factors Influencing Slow Crack Growth

There are several mechanisms by which a subcritical flaw may grow in service. For example, hydrogen embrittlement, stress-corrosion cracking and fatigue are well known causes of slow crack growth. Spontaneous strain-aging is another, not generally recognized mechanism of subcritical crack growth. Each of these mechanisms has been under study at Aerojet-General. Research on these phenomena is conducted by means of fracture test specimens containing fatigue pre-cracks. Crack growth is monitored both by crack-opening displacement and by stress-wave emissions.

(1) Slow Crack Growth in Fatigue

Research in fatigue was conducted for NASA⁽¹³⁾ to investigate selected variables affecting fatigue crack growth. Both D6aC and 18% nickel maraging steel bend and center-notch tensile

specimens were employed during this program. Through the tests conducted, it was possible to monitor incremental fatigue crack growth and determine differences in stress-wave emission measured with SWAT as a function of varying test conditions, material conditions, and heats. Relationships between the number and amplitude of the stress-wave emissions (SWE) versus crack growth rate were established. It was verified that fatigue crack growth is discontinuous, and for any given cycle the peak stress-wave emission and amplitude is generally associated with maximum load--with the largest growth occurring at peak load. There were also significant differences in SWE as a function of loading conditions. For example, in D6aC steel heat treated to high toughness (Table II), low-cycle high-stress-intensity

Table II Tensile and Impact Properties of D6aC 0.125-in. Sheet

Heat No. and Purpose	Treatment ^(a) of	Ultimate (Ksi)	.2% Offset Yield (Ksi)	Precracked Charpy (W/A, in-lb/in ²)
3960907 fatigue tests	1550/400	204.3 209.0	152.4 157.1	148 154
	Avg.	206.7	154.8	151
	1750/1100	226.5 226.8	202.1 203.0	1078 1110
	Avg.	226.7	202.6	1094
3951282 hydrogen and/or H ₂ O tests	1550/400	196.0 185.0	139.0 128.0	108 132
	Avg.	190.5	133.5	120
	1750/1100	235 225	212 207	1055 1023
	Avg.	230	209.5	1039
(a) Austenitized at 1550°F for 1/4 hr, air quenched and tempered at 400°F for 2 hr; austenitized at 1750°F for 1/2 hr, air quenched and tempered at 1100°F for 2 hr.				

fatigue produced as many as 64 stress waves in a single load cycle. Quantitative measurements of SWE amplitude were made of individual cycles, and the crack growth in each cycle was determined by micrometer-microscope measurement of the fracture-surface markings such as those seen in Figure 18.

Detailed data are presented in Table III. From these data, three observations are considered to be significant. First, as the crack-growth rate increases, generally the number, amplitude, and summation of the SWE increases. Secondly, the first cycle of each new applied stress level was attended by a larger number, amplitude, and summation of SWE than the following cycles at the same applied stress level. This correlates with the fact that generally there was also a larger crack-growth rate in the first cycle of each higher applied stress level. For example, in cycle 34, the number,

Fig. 18 Fracture Surface Marking (Low-Cycle High-Stress-Intensity Fatigue in High-Toughness D6aC Steel). Magnification, approximately 8 X.

Table III Fatigue Crack Growth Data from 8 x 24-in. D6aC Panel (Specimen 1750-2)

Cycle No.	Crack Growth Rate (d2c/dN) (in/cycle)	SWE Count	Max. SWE (g)	SWE.g (g)
20	0.0117	8	0.30	2.1
21	0.0134	6	0.26	1.3
22	0.0121	7	0.26	1.3
23	0.0114	9	0.28	2.3
24	0.0126	4	0.30	1.1
25	0.0112	5	0.28	1.0
26(a)	0.0354	18	0.60	5.8
27	0.0264	18	0.28	5.2
28	0.0263	20	0.43	5.4
29	0.0245	13	0.33	4.4
30	0.0246	18	0.30	5.2
31	0.0231	9	0.33	2.3
32	0.0235	18	0.38	5.5
33	0.0273	13	0.61	4.2
34(a)	0.0568	33	0.90	10.2
35	0.0415	23	0.36	5.8
36	0.0392	26	0.42	6.5
37	0.0507	31	0.47	8.8
38	0.0475	30	0.40	6.9
39	0.0494	27	0.90	7.0
40	0.0520	32	0.72	8.1
41(a)	0.1042	40	1.0	13.8
42	0.1243	48	0.70	16.2
43	0.1671	64	0.70	24.1
44	Failure	45	1.0	33.0

(a) Increase in applied load

largest amplitude, and summation of SWE are 33, 0.90, and 10.2, respectively. These correspond to average values in the next six cycles at the same applied stress level of 28, 0.55, and 7.2. Significantly, the crack growth rate in cycle 34 was 0.0568 in./cycle as compared to an average of 0.0467 in./cycle in the subsequent six cycles.

When the crack growth associated with each load cycle was measured in the fracture surfaces and plotted against the summation of the stress-wave amplitudes, a direct proportionality was indicated as shown in Figure 19.

Fig. 19 Summation of Stress-Wave Emission vs Crack Growth Rate (Two Conditions of D6aC Steel).

In the D6aC steel heat treated to a low-toughness condition, low-cycle high-stress intensity fatigue was associated with dormant periods ranging from two to three cycles to hundreds of load cycles, followed by macroscopic growth in a single load cycle. In the last few cycles of load before failure in the low-toughness condition, there were relatively few stress waves as a precursor to failure. Consider the test results of a typical specimen cycled at progressively higher stress levels until failure (Figure 20). At a stress intensity factor of 52 ksi-in.^{1/2} ($K/K_c = 0.63$), the first load cycle was attended by numerous small SWE's starting at approximately 50% of maximum load, with three large stress waves and a burst of intermediate-amplitude SWE's at maximum load. In the second and third load cycles, no SWE's occurred on rising load; however, there were two or three stress waves at maximum load in each of these cycles. During cycles 4 through 8 at a slightly reduced stress intensity level, no

Fig. 20 Stress-Wave Emissions and Load vs Time
(First Three Cycles, D6aC, Specimen 1550-1, 900°F Temper).

SWE's were observed. In cycle 9, at a higher stress-intensity level (78 ksi-in.^{1/2}), the cracking resumed with numerous small SWE's at maximum load and one medium-to-large SWE. In cycle 10, unlike cycles 2 through 9, there were small SWE's on the rising load; at maximum load the specimen fractured.

The relationship between Σ SWE·g and crack growth also has been observed in other materials in single cycles of rising load to failure. For example, in SEN tensile tests of 6Al-4V titanium heat treated to 174 ksi ultimate tensile strength, there was an increase in number and amplitude of stress waves with increasing stress-intensity level, providing ample time to locate the defect source and release the load if an automated SWAT system were in use. When the sum of the stress-wave amplitudes over a given crack-growth increment was plotted against the amount of crack growth occurring in that increment (at a 0.1-g sensitivity level), a direct proportionality was indicated (Figure 21).

Research on fatigue in HY-80 steel was conducted for the David Taylor Model Basin⁽¹⁴⁾ to determine the feasibility of monitoring fatigue crack growth in HY-80 weld and heat-affected base metal by stress-wave emission. Bend specimens (0.75 in. thick by 2 in. wide by 24 in. long) fabricated from a weldment supplied by the U.S. Navy containing a transverse weld were tested utilizing a part-through-surface-crack and loaded in pure bending. Various sizes and shapes of initiating flaws located in both the weld metal and heat-affected zone were investigated. The results obtained showed that SWAT could be employed to monitor subcritical fatigue crack growth in HY-80 weldments. Relationships between applied stress intensity and crack-growth rate were established using analytical techniques

Fig. 21 Stress-Wave Emission vs Crack Growth in 6Al-4V Titanium (174-ksi Tensile Strength SEN Tensile Specimen 0.10-in. Thick).

and equations developed at Aerojet-General to account for the varying surface and in-depth stress intensity factor which occurs in bending when testing a PTC-type flaw.

Although a quantitative determination of the relationship between stress-wave emission (SWE) and fatigue-crack growth rate was not obtained, the data indicated a possible direct proportionality between the summation of SWE amplitudes and the amount of crack growth per cycle.

(2) Environmentally Induced Slow Crack Growth

Research is being conducted for the Air Force Materials Laboratory⁽¹⁵⁾ to investigate the metallurgical nature of slow crack growth, the mechanisms by which it occurs, and how it is affected by hydrogen level, microstructure, environment, rate of loading, test temperature, and stress level. The slow-crack-growth mechanisms under investigation are hydrogen embrittlement, stress-corrosion cracking, and spontaneous strain aging.

(3) Hydrogen and Stress-Corrosion Cracking

Figure 22 illustrates the setup for testing in an aqueous environment. An aluminum cup is attached to the single-edge-notch tensile specimen with a pliable adhesive so that the fatigue precrack is immersed in water during testing. The holes are for pin

Fig. 22 Single-Edge-Notch Tensile Specimen (Environmental Test Setup).

loading. The crack-opening displacement gage is clearly shown. The accelerometer, not shown, is attached directly to the specimen.

To illustrate the findings of this investigation consider the results obtained from a D6aC test specimen, loaded to 6 kips ($K/K_c = 0.056$) and held at load, in air, for approximately 3 hr with no SWE and no detectable slow crack growth indicated by crack-opening displacement gage (Table IV). While still under a 6-kip load, a drop of distilled water was placed on the side of the test piece at the tip of the edge notch. In approximately 1 sec, a burst of SWE was recorded ranging in amplitude from approximately 0.1 to 0.3 g. The burst lasted for approximately 11 sec and contained 11 emissions.

A second burst occurred about 24 sec later, ranging in amplitude from approximately 0.3 to 0.5 g. The duration of the burst was 5 sec and contained five emissions. Then a single emission occurred approximately 100 sec later.

A third burst occurred about 524 sec later, ranging in amplitude from approximately 0.4 to 0.8 g. The duration of the burst was 64 sec and contained eight emissions. A single emission occurred 59 sec later.

Table IV Stress-Wave Data from an Environmental Test of D6aC Steel
(Hydrogenated, Plated, Baked and Tested in Distilled Water)

TIME IN SECONDS AFTER ADDING H ₂ O ELAPSED (AFTER H ₂ O)	INTERNAL BETWEEN SWE	SWE AMPLITUDE (g UNITS)
0		
1 00	1 00	0 18
1 39	0 39	0 20
1 46	0 07	0 24
2 15	0 69	0 26
2 24	0 09	0 10
2 49	0 25	0 16
2 52	0 03	0 12
2 82	0 30	0 24
2 88	0 06	0 18
9 20	6 32	0 12
11 57	2 37 Range 07 6 32 Avg (11) 1 05	0 24 Range 10 26 Avg (11) 0 18
36 05	24 48	0 49
36 39	0 34	0 39
37 19	0 80	0 37
38 29	1 10	0 32
41 57	3 28 Range 34 3 28 Avg (4) 1 38	0 28 Range 28 49 Avg (5) 0 37
142 55	100 98	0 16
666 55	524 00	0 43
666 58	03	0 75
691 43	24 85	0 83
702 41	10 98	0 47
721 01	18 60	0 55
721 03	0 02	0 71
711 25	1 22	0 59
731 33	9 08 Range 03-24 85 Avg (7) 9 25	0 83 Range 43 83 Avg (8) 0 61
790 33	59 00	0 32
1030	UNSTABLE CRACK GROWTH FRACTURE	

A fourth burst of SWE occurred 1030 sec after the water was added; this burst resulted in fracture of the test specimen. Thus, by means of the SWE, we are able to follow step-by-step, with complete detail in each step, the subcritical crack growth induced by stress-corrosion cracking.

Figure 23 illustrates the stress-wave emission that occurs in D6aC steel when heat treated to low toughness. Each line represents a stress wave, with amplitude plotted on the ordinate scale and time of occurrence on the abscissa. Only the stress waves larger than 0.01 g are shown in the bar graph. Note that as time in the environment increased, there was an increase in the rate of stress-wave emission. At final instability there was a marked increase in both rate and amplitude of the stress-wave emission. Only the larger stress waves could be shown in the bar graph; in the written report, 182 stress waves were recorded for a hold period of 77 min, or an overall average interval between stress waves of 27 sec. When all the stress waves were taken into account, an excellent correlation was found between the crack-growth rate as measured by crack-opening displacement gage and the rate of stress-wave occurrence.

Figure 24 shows the correlation between SWE and crack-growth rate in water. In this figure we have a comparison of the crack-growth rate as measured by the calibrated crack-opening displacement gage and the rate of SWE. Note the excellent correlation between crack-growth rate and the number of stress waves recorded per unit time. The abrupt change in slope occurring after approximately

Fig. 23 Distribution and Amplitude of Stress Waves at Sustained Load in D6aC (Austenitized at 1550°F, Tempered at 400°F, Hydrogenated, Baked at 300°F/3 hr and Held in Distilled Water. Only stress waves >0.01 g are shown.

Fig. 24 Stress-Wave Emission and Crack Growth Rate vs Time Toughness D6aC Steel Specimen Austenitized at 1550°F, Tempered at 400°F, Hydrogenated, Baked for 3 hr at 300°F and Tested in Distilled Water).

68 min at load corresponds to the onset of rapid crack growth or crack instability. This abrupt change in slope has now been observed in a number of materials and is a most important observation from a non-destructive inspection standpoint, and provides at least one piece of information for use in an on-line SWAT system monitoring the degree of flow criticality.

The data of Figure 25 are for D6aC material that has been deliberately hydrogenated and then tested in water. The

Fig. 25 Stress-Wave Emission Rate and Crack Growth Rate vs Stress Intensity Factor (D6aC Steel Austenitized at 1550°F and Tempered at 400°F).

applied stress-intensity factor is plotted against (1) rate of crack growth as measured by the crack-opening displacement gage and (2) rate of stress-wave emission as measured by the accelerometer. Note the parallel trends in the two sets of curves, providing further evidence that the data obtained from the accelerometer are in fact a measure of crack growth. With regard to the effect of hydrogen and water in

promoting slow crack growth, when high-toughness, hydrogenated D6aC steel was tested in water, it cracked at approximately the same rate as hydrogenated material tested in air; thus, it appears that hydrogen is the controlling mechanism. On the other hand, when low-toughness D6aC steel was tested, the combined effect of hydrogenation and testing in water produced a higher crack-growth rate than obtained from either hydrogenation or aqueous environment tested alone, indicating a synergistic effect.

(4) Spontaneous Strain Aging

Under the Air Force contract⁽¹⁵⁾ spontaneous strain aging was also investigated as a mechanism of slow crack growth in D6aC steel. Single-edge-notch tensile specimens were exposed to moderately elevated temperature (400 to 500°F) under sustained load. Compared to hydrogen embrittlement and stress-corrosion cracking, the embrittlement at elevated temperature was attended by relatively few movements of the crack front. Light and electron fractography confirmed that the large stress waves that punctuated the periods of sustained load were associated with gross movement of the crack front. At the start of the holding period the time between stress waves (Δt_s) was short, with Δt_s increasing with time at load until finally a steady-state condition was reached where the secondary incubation time was relatively constant (Figure 26). After the D6aC heat treatment which effectively solutionized the carbides (1750°F for 1/2 hr), the steady-state secondary incubation time was approximately 80 sec, as

Fig. 26 Crack-Growth Rate vs SWE Secondary-Incubation Times (D6aC Steel Tested at 500°F).

compared to 1650 sec in the heat-treat condition with massive carbides (1550°F for 1/4 hr). The crack growth rate also varied by a factor of 20 for the two heat-treat conditions. Thus, it appears that the embrittlement is a function of the amount of carbon in solid solution; with an effective solutionizing heat treatment, the kinetics of both secondary-incubation time and crack-growth rate suggested the embrittling mechanism to be carbide precipitation. Consistent with other phases of the research, a relationship was found between crack growth rate as measured by crack opening displacement gage and rate of stress-wave emission in the elevated-temperature testing (Figure 27).

Fig. 27 Stress-Wave Emission Rate and Crack Growth Rate vs Elapsed Time (Secondary Incubation Time—D6aC Specimen X-43 Tested at 500°F).

III. SWAT SYSTEMS

1. Special-Purpose System

Based upon the success of the applied SWAT technology Aerojet has currently developed two field-application SWAT systems which overcome the difficulties involved in using magnetic tape recorders and posttest data analysis techniques. The first SWAT system, shown in Figure 28, is a special purpose system designed for pressure vessels of simple geometry and primarily intended for use in locating propagating defects during manufacturing pressure tests or service use. This SWAT system consists of signal conditioning, analog-to-digital converters, digital logic network, data output, and visual-graphic display.

Fig. 28 SWAT System--Special Purpose
(Optical Triangulation Display).

Stress-wave sensor inputs are conditioned and then converted from analog to digital values which are processed through the logic network and the data and results displayed. The graphic display for location of the source of stress-wave emission consists of a triangulation technique using circles of light whose projected radius is equivalent to the difference in time of stress-wave arrival multiplied by the velocity of the stress-wave in the pressure-vessel material.

This system can be preset to perform a limited number of pre-established reactions upon sensing stress-wave emissions. These reactions range in response from merely signalling that a stress-wave has been sensed to controlling the applied pressure or loading system. A pressure vessel production facility is now using this SWAT system during the qualification hydrostatic proof pressure test of each of its manufactured products.

2. General-Purpose System

A second SWAT system encompasses a much greater degree of sophistication. This system which is capable of accepting as many as 32 sensor inputs also conditions and converts the analog signal into digital information. The digital information can then be handled in either of two modes: the first is a data logging method which presents all the information processed through the digital logic networks to a visual nixie-tube display and simultaneously records this data and the time at which it occurred on punched paper tape and as a typewritten

record. The primary use of this data logging method is for long-term surveillance of noncritical structural components, time dependent material properties tests, and other applications where the using agency does not require an immediate responsive action.

The second mode in which the stress-wave data may be handled is more universal in nature. In this mode all of the digital stress-wave data and any other pertinent information such as temperature, pressure, load, strain, etc., is furnished internally to a general purpose digital computer. This general purpose computer can be programmed to operate on the data in any manner desired by the user. Triangulation programs for arbitrary geometry test specimens have been developed so that flaw location can be automatically furnished as data output. Through this system all possible data handling techniques can be automatically accomplished and displayed, in real-test-time, so that decisions or actions may be taken. The computer can also be scheduled to make decisions and react, responsively, to any pre-established situation, on applications where long-term surveillance of critical components is of primary importance.

3. Current SWAT Applications

The ever broadening amount of knowledge relating stress-wave emissions to flaw growth/failure processes has substantially increased the range of SWAT applications and materials research. In order to provide the reader with a more complete picture of the applicability of this technique we have briefly itemized some of our current activities.

For lower strength metals:

- fatigue tests of pressurized piping systems;
- natural gas transmission pipelines;
- pressure vessels, heat exchangers, and boilers
for nuclear reactor and chemical processing;
- gas pressurization storage-bottle farms;
- biaxial fatigue tests of naval ship-hull materials.

For higher strength materials:

- determination of the residual static strength of
flawed structures;
- systems applications for research laboratories;
- detection and location of defects propagating
during pressure qualification tests of aerospace
pressure vessels.

IV. APPLICATION OF SWAT TO REACTORS

The application of SWAT technology to reactor problem areas is currently within the state-of-the-art. The simplicity of application begins with the use of the technique during the pressure qualification tests required of all pressure vessel manufacturers and ends with the complexity of application to detect malfunctions in the operating core of the reactor. Room temperature, static pressure tests provide the optimum time for the application of SWAT from both an accessibility and operational (extraneous) noise standpoint.

Environmental conditions (radiation, temperature) impose a much more severe test of the capability of the technique. Much effort is being devoted to solving these environmental problems, particularly the effects of radiation and temperature on the sensor. If these environmental difficulties can be met, then stress-wave surveillance of any portions of the reactor system can be accomplished.

Aware of these problem areas, we have been applying our background of instrumentation knowledge regarding radiation hardening and temperature conditioning, acquired in the nuclear rocket program, towards developing acceptable stress-wave sensors for use in nuclear power plants. Radiation hardened and temperature conditioned accelerometers have been utilized to 500°F at a total gamma exposure of 1.1×10^{10} Ergs/gm(c) and a total fluence of 1.8×10^{16} neutrons/cm² ($E > 0.9$ Mev).⁽¹⁶⁾ Other applications have already extended accelerometers for use at -423°F in temperature and up to 20,000 psig in pressure capability.

For use in situations where higher sensitivity and wider frequency response may be desirable, crystal materials for use to approximately 1150°F are being evaluated. Higher frequency responses may be required when the technique is employed in operational systems with a large amount of low frequency noise. A key feature of stress-wave emission, its transient nature, allows us to receive signals considerably higher in frequency than the majority of mechanical system noise. Thus, selectively filtering the raw data will significantly improve the signal-to-noise ratio, enhancing the detection of stress-wave emission.

An important problem in connection with reactor containment vessels is the progressive embrittlement due to radiation damage. More specifically, there is an urgent need for a nondestructive inspection technique which will provide some measure of the damage at any given point in time. The fact that D6aC low-alloy steel, heat treated to two levels of toughness (see Table II) behaves so differently as measured by SWAT, indicates that progressive embrittlement due to radiation damage may be readily assessed by this method.

V. OTHER APPLICATIONS

The application of an on-line SWAT system to an operational reactor facility can provide some interesting sideline benefits. A data-analysis technique, much in discussion today, is termed mechanical signature analysis. This technique had its inception with such uses as determining when rotating machine parts, such as roller bearings, were approaching failure by progressively comparing data records made at varying times of the in-service life, until distinctive changes were correlated with observable decreases in performance.

In a similar manner, stress-wave sensors properly applied and analyzed have great potential as a method for detecting other system malfunctions. The working of joints and the leakage of fluid have all been readily detected by SWAT during tests. Other investigators have shown these same techniques capable of detecting cavitation of pumps,⁽¹⁷⁾ nucleate boiling,⁽¹⁸⁾ and detrimental vibratory motion.

With the on-line application of a SWAT system, automatic mechanical-signature-analysis techniques can be readily accomplished.

The SWAT system can be activated to conduct these signature-analysis techniques at preselected increments of the operational facilities life and provide these data directly to the cognizant personnel. This application of the SWAT system may, of itself, be a significant achievement.

VI. CONCLUSION

The stress-wave analysis technique is one of the most exciting developments in the engineering field within recent years. The potential utilization of the technique as a method for detecting incipient failures has been demonstrated on materials ranging from elastomers to glass. With SWAT one cannot only determine if there is a growing flaw, but also locate the flaw by triangulation. In most structural metals, as the crack increases in length, there is an increase in the rate and amplitude of SWE that presages failure.

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Detection, Location, and Characterization of Flaw Growth in Metals Using Acoustic Emission Methods

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ABSTRACT

Battelle - Northwest in February of 1966 initiated a program under AEC/DRDT sponsorship to nondestructively detect incipient failure of reactor pressure piping using the acoustic emission generated by fracture, cleavage, dislocation unpinning, twinning, etc. of single or multiple grains of the piping material. The purpose of the program is fourfold; namely (1) the detection of acoustic emission against the noise background of the operating reactor, (2) the location of the point of origin of the acoustic emission, (3) the characterization of the acoustic emission by the type of failure or flaw growth mechanism in the metal, and (4) prediction of the time of failure.

The Detection of Acoustic Emission Against A Reactor Noise Background

One approach to the detection of the acoustic emission against a noise background is to sense the acoustic emission in frequency bands where the noise is not present. Hutton et al.¹ have shown that the operating noise spectrum of the Hanford "N" reactor, as measured on the back face, is all below 20 KHz, and that a high pass, 4th order filter with a cutoff at 20 KHz effectively eliminated the noise.² Hutton et al.³ have also shown that noise, generated by boiling in pressurized containers, 680°F at 2600 psi, can be again effectively eliminated using a 20 KHz high pass filter. The procedure used to date for elimination of noise background is to use a fourth order, 20 KHz high pass filter to eliminate the noise and to allow detection of the acoustic emission signal in the frequency band of 20 KHz to 300 KHz.

*Work sponsored by the AEC under Contract AT-(45-1)-1830 at the Pacific Northwest Laboratories of Battelle Memorial Institute.

Ord² has shown that a brittle single edge notch tensile specimen will generate acoustic emission in the 700 to 800 KHz using equipment having a 1.5 MHz bandwidth. It is thus recognized that upgrading of the present detection equipment is required.

Flaw Location

The location of the point of origin of an acoustic emission signal is a straight forward application of triangulation. The only problem is the accuracy to which triangulation can be accomplished. The accuracy is highly geometry dependent. In the case¹ where a "free field" propagation condition exists, the location error to a fraction of a percent of the distance between transducers can be achieved using three transducers, and that the error is reduced by $1/\sqrt{n} - 1$ using four or more transducers. Generally in practice "free field" conditions do not exist and this is especially true in piping due to resonances, moding, multiple paths, etc. In general Ord¹ has found that for a three transducer configuration that error distances of 5% of the distance between transducers can be achieved in piping.

Signal Characterization

Attempts to correlate the number of acoustic emission events with the number of failure events in a metal has been shown by Hutton et al.⁴ to be somewhat possible. Tests performed on a single edge notch tensile specimen of D-2 tool steel resulted in 10^6 acoustic emission events/in.² of surface generated for 10^6 cleavages/in.² and 10^9 ductile dimples/in.² as determined by replication techniques. In general ductile mechanisms (80 - 100 KHz) are lower in frequency while brittle mechanisms are much higher, ranging up to at least 800 KHz.

Failure Prediction

Hutton et al.⁴ have shown that during burst testing of 24 inch diameter, 1.6 inch wall, pressure cylinders 12 feet long, reliable prediction of failure was possible 11 minutes prior to failure, which tends to verify acoustic emission as a reliable method of detecting incipient failure.

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State of the Art of Nondestructive Testing for Service and Postoperation Examination of Reactor Pressure Vessels*

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Introduction

Nondestructive examination is generally performed on non-nuclear thick-walled pressure vessels during shutdown for maintenance. The principal method has been visual examination with occasional improvements in sensitivity or accessibility from such aids as liquid penetrants and optical systems. Visual examination is also a vital procedure for nuclear pressure vessels. However, for several reasons additional nondestructive testing methods are needed to assure the continued integrity of the nuclear pressure vessels after extended service. These tests may be performed during reactor operation or during intervals of shutdown for core change, maintenance, or other reasons.

During the fabrication of the vessels, a number of nondestructive examinations are performed on both the component materials and the completed vessel.¹ The test methods include radiography, ultrasonics, liquid penetrants, and magnetic particle examination. Since the components of the vessel are examined prior to assembly, particular attention is given to the welds of the vessel when it is inspected in its final form. Radiography is the major test method for the final weld examination although increasing use is being made of ultrasonics.

Current Practice

A number of reactor operators were contacted to determine current practice for the nondestructive examination of nuclear pressure vessels. In general, the

*Research sponsored by the U.S. Atomic Energy Commission under contract with the Union Carbide Corporation.

responses were similar in respect to the designation of methods considered to be applicable and necessary. In all cases emphasis was on visual inspection. Occasionally, liquid penetrant inspection of a portion of the cladding within the vessel has been performed. In most cases optical aids such as replication, borescopes, periscopes, or television cameras have been employed where direct viewing was not possible. Since these tests are sensitive only to discontinuities open to the accessible surface, consideration is being given to supplementary examination methods to detect the generation or propagation of harmful defects in other areas of the pressure vessel.

Candidate Examinations

There are a number of possible approaches for pressure vessel examination which have been proposed. Some of these methods have been applied to large components; others are still in various stages of development. Most of the suggested supplementary methods involve some form of acoustic phenomenon, although radiography and other methods have been considered.

On some vessels certain limited portions, such as the flange region or nozzle weldments, may be sufficiently remote from the reactor core that low background radiation could allow conventional manual ultrasonic examination. However, all areas of concern cannot be covered by manual techniques so supplementary examination systems must be considered. For convenience in discussion, the acoustic methods will be divided into two major categories: (1) those in which mechanical scanning of the acoustic probes is performed; and (2) those in which the probes are fixed upon the pressure vessel.

Mechanical Scanning System

Scanning systems have been applied to reactor systems or similar configurations with sufficient success to demonstrate the feasibility of such an approach. For example, an ultrasonic scanning unit was designed and fabricated to allow immersed ultrasonic examination of the pressure vessel from the PM-2A nuclear power plant after the vessel was shipped to the Nuclear Reactor Test Station, Idaho.² The test unit rotated about a center shaft on the vessel axis to allow evaluation of the entire vessel from the interior. An additional fixture was built to examine the primary nozzles.

A number of devices were developed for remote maintenance and inspection of the Homogeneous Reactor Test core vessel and accessory items. Included was an ultrasonic system for measuring the wall thickness of the core vessel.³ Prototype mechanisms were developed for ultrasonic examination of nozzle welds for the Elk River Reactor pressure vessel.⁴

Equipment is being developed in several laboratories to attach to the wall of a pressure vessel to perform mechanized ultrasonic scans. Similar work has been done for the examination of submarine hull weldments.

There are a number of problems which must be solved to allow worthwhile mechanized ultrasonic scanning. These problems include the effect of the overlay cladding on the results obtained from the examination, establishment of bench marks for orientation, and attainment of reproducible records for comparing with subsequent evaluations.

Fixed Probe System

An alternative to the ultrasonic scanning system is the use of probes which are fixed at key positions to monitor areas of concern. The system can

be active or passive. An active system initiates the sonic energy, detects it, and denotes a change as a result of the energy being transmitted into and through the specimen. A passive system detects the energy which may have been released by other conditions. Since extensive discussion on acoustic emission, the passive system, has been presented in this session, no further elaboration is necessary.

Active systems have been used in which pulse-reflection transducers were applied in a fixed position to monitor crack growth during cyclic loading of fatigue specimens, weld seam sections from large-diameter pipes,⁵ or even complete pressure vessels.⁶ Unfortunately, in some cases, failure did not occur at the area being monitored. Another potential active system which has had limited application to large pipe and rocket motor cases uses multiple transducers or "wave directors" mounted directly to the specimen surface for use as transmitters and receivers.⁷ Continuous, mechanized adjustment of the angle of the transmitters and receivers allows scanning of the specimen by beam intersection techniques despite the fixed position of the probes. A wave guide action in the wall of the specimen and frequency analysis techniques optimize the transmitter-receiver efficiency. Problems for the fixed-probe systems are similar to those for the scanning systems. Use of fixed probes for evaluation during operation will require assurance that the components of the test system can tolerate the radiation and temperature environment.

Other Test Methods

Other tests methods such as radiography, magnetic testing and penetrants should receive continuing consideration as technique improvements make them potentially applicable to the post-operation evaluation of pressure vessels.

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Nondestructive Measurement of Surface Residual Stress

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INTRODUCTION

The term residual stress is the name given to the stress that exists in a material when all the external forces have been removed. If these stresses are uniform over large distances they are called macroscopic. If these stresses vary from grain to grain or exist in areas near dislocations, they are referred to as microscopic. These internal stresses can be due to a number of conditions such as nonuniform cooling or quenching, phase transitions occurring during processing, mismatch of components constrained mechanically prior to welding or joining, and mechanical working, rolling, drawing, machining or grinding. Experimentation has shown that the total residual stress on the surface of a machined part is the combined effect of the machining or grinding operation and the accompanying heat involved in the process of removing the material. Much literature is available on the effects of various machining operations and machining modes in producing stress. This literature is controversial and even contradictory. However, the residual stress layer is generally confined to a relatively shallow depth within 0.005 - 0.010 inches of the surface.

The type and magnitude of the stress is one of the fundamental factors which determines the mechanical properties of solids when being deformed. For example, a notch or groove in a specimen produces nonuniform cross-sectional distribution of stresses below the yield strength and the nonuniformity in stress distribution is also retained beyond the yield strength, thus preventing plastic deformation. The deleterious effects of residual stresses are warpage during machining or grinding, lack of dimensional stability with time, and reduced resistance to premature failure by fatigue or stress corrosion. A quote from Dr. Weck(1) emphasizes the importance of residual stress:

*Mr. McGonnagle was an invited speaker but was unable to attend, and his paper was not presented during the conference.

"...the location of a defect in a field of residual stresses, whether tensile or compressive, may influence the effect of the defect both on fatigue strength and brittle fracture strength..."

The residual stresses are superimposed on the load stresses, and if the combined stresses exceed a certain critical value they may cause failure of a component.

Various destructive techniques have been used to measure residual stress all making use of the phenomena of strain relaxation. The strain relaxation techniques are based on the assumption that when the external load is removed from a metal specimen it behaves in a purely elastic manner, even if it has undergone plastic deformation. The elastic-strain release that occurs is measured when a specimen is cut into pieces or layers are removed from the specimen. The stress is then calculated using equations derived from the elastic theory of solids (2-4). The assumption, used in all destructive methods of measuring residual stresses, that the destruction of the continuity of a body in which residual stresses are present give rise to elastic strains alone is not entirely correct. Because removal of layers of metal can cause elastoplastic strains, from which incorrect values of residual stresses will be given by calculations using the usual formulas.

Making the destructive analysis by removal of material is generally difficult because on the average, stresses can only be computed in layers or portions which have been removed from the specimen. This requires removing a large number of extremely thin layers which in turn results in an expensive and time consuming analysis often requiring the use of a computer. In making a detailed study of destructive techniques of measuring residual stress, it was found that questionable assumptions were made, there was no uniformity in technique, there was a lack of suitable standards and there was great variation in the results.

The information desired from the destructive and nondestructive test is type of stress (tensile or compressive), magnitude of the residual stress and the stress pattern (the variation of stress with depth in the material).

Although there are many factors which determine the final configuration of surface residual stresses in a machined piece, some general features of the residual stresses have been observed by various investigators and reported in the literature. No discussion will be presented in this paper on the effect of the specific tools, shapes, sharpness of the cutting edges, and modes of cutting in relation to the induced total residual stresses, but a series of figures in references (5-11) illustrate the residual stress patterns in various materials for different cutting parameters employed. Figures 1 and 2 show that the maximum residual stresses (either compressive or tensile) are induced at the surface of the metal; then the type of stress changes to balance the stresses introduced at the very near surface.

NONDESTRUCTIVE METHODS

The nondestructive testing methods are, in general, based upon the stress-dependent properties of materials. By measuring these stress-dependent properties, the changes in the properties can be related to the magnitude of the stresses existing in the material. Nondestructive methods are more desirable in measuring residual stresses because they do not damage the material.

Nondestructive methods used or proposed to measure residual stress include: (a) mechanical strain gages (4), (b) photoelastic coatings (12),

Fig. 1 Residual Stress Resulting from Grinding Operations on a Workpiece.

Fig. 2 Comparison of Residual Stress Resulting from Parameter Change in Milling Operation.

(c) photoelastic effect (13), (d) acoustoelastic effect (14), (e) magnetic (15), Barkhausen (16), electrical (17), and (f) X-ray (18-20). However, none of the techniques have been developed beyond the laboratory experimentation stage except the X-ray method. The X-ray method of measuring residual stresses has been in existence for some time, and the techniques developed have proven successful from a practical point of view for certain selected applications.

X-RAY METHOD

X-ray diffraction is now the most extensively used method for determination of residual stress. The X-ray method is based on measuring the change in the crystal lattice spacing caused by the residual or applied stress in the specimen. Only the first few atomic layers of the surface are involved. When a polycrystalline piece of metal is deformed elastically the lattice plane spacings in the constituent grains change from their stress-free value to some new value depending on the magnitude of the applied stress. This new spacing is essentially constant from one grain to another for any particular set of planes. If the metal is deformed plastically, the lattice planes usually become distorted in such a way that the spacing of any particular set of planes varies from one grain to another or from one part of a grain to another.

When a monochromatic X-ray wave is reflected from the atomic planes, the diffraction of X-rays satisfies the Bragg equation:

$$n \lambda = 2d \sin \theta \quad (1)$$

where λ = the wavelength of incident beam

θ = the angle between incident or reflected beams and surface of reflecting lattice planes

d = the lattice spacing

n = the order of reflection ($n = 1, 2, 3, \dots$)

Equation (1) shows that if the wavelength of the X-ray is known, the interplanar spacing, d , can be determined by measuring the angle θ . The residual stresses are then deduced from the lattice strains by using mathematical relations deduced from the theory of elasticity. In deriving the equations for calculating the stress, it is assumed that the isotropic elastic constants can be used. This is not absolutely correct because the stiffness of the grains varies with orientation. The precision in measuring the interplanar spacings depends on the precision in measuring θ . To evaluate the effective errors, the Bragg equation is differentiated with respect to θ ;

$$\frac{\Delta d}{d} = \cot \theta \cdot \Delta \theta \quad (2)$$

This equation shows that the larger the angle θ , the better the precision that can be obtained. Since it is difficult to measure θ to better than 0.005° , diffraction angles of 60° and preferably over 70° are used to select X-ray radiation wavelengths for the particular material to be investigated. In most cases the optimum conditions cannot be used because of adverse X-ray scattering by the specimen. When

this condition occurs, other diffraction planes must be chosen at the expense of the precision of analysis.

A uniform macrostrain causes a shift of the diffraction lines to new positions. Plastic deformation causes a broadening of the diffraction lines. In this discussion we will be concerned with the line shift due to uniform strain. From this shift the strain may be calculated and, knowing the strain, one can determine the stress present, either by a calculation involving the mechanically measured elastic constants of the material, or by a calibration procedure involving measurement of the strains produced by known stresses. The theoretical developments necessary for the treatment of lattice strains and their relation to stresses follow from the classical theory of elasticity and are given by Barrett (20). For an isotropic solid under homogeneous deformation, it can be shown that, for uniaxial stresses

$$\epsilon_{\Psi L} = \frac{\sigma_L}{E} [(1 + \nu) \sin^2 \Psi - \nu] \quad (3)$$

$$\epsilon_{\Psi T} = - \frac{\nu \sigma_L}{E} \quad (4)$$

where σ_L is the macrostress acting in the longitudinal direction; $\epsilon_{\Psi L}$ is the lattice strain in the plane defined by the surface normal and the longitudinal direction at an angle, Ψ , from the surface normal; $\epsilon_{\Psi T}$ is the similar strain in the transverse plane; E is Young's modulus; and ν is Poisson's ratio.

Determination of the stress by X-rays is usually not done by direct calculations of the lattice strains but by determining the stress according to the following equation

$$\sigma_{\Phi} = K \Delta\theta \quad (5)$$

when $\Delta\theta$ is the X-ray peak position shift between the normal ($\Psi = 0$ deg) and oblique ($\Psi = \Psi$) directions and Φ is any surface direction (19) and K is a material constant. X-ray technique coupled with the layer removal technique permit approximate determination of the depth of the stress layer. This X-ray technique is used for determining macrostresses. In X-ray stress analysis the beam width varies from 1.0 to 0.1 cm. If the stress varies over the beam width it would cause line broadening and be classified as a microstress.

Two general techniques are used in the recording of diffraction patterns: (a) photographic and (b) X-ray diffractometer. In both techniques the preceding analysis assumes isotopic material. In general, this assumption is not valid for the individual grains of the specimen even when it is valid for the specimen as a whole. This consideration, together with the fact that a diffracted beam comes from only those grains having a certain orientation in the specimen, has given rise to much uncertainty concerning the correct values to use for macroscopic or average values of Young's modulus and Poisson's ratio (21,22).

An approximate average sensitivity of macroscopic strain measurement is given in Table I. It should be noted that these values are all determined from the average values of lattice residual strain in the specimen. The chemical composition of the specimen affects the test results (21).

TABLE 1
SUMMARY OF RESIDUAL STRESS MEASURING METHODS

	Film Technique	<u>X-Ray Diffraction</u> Diffrac- tometer	Ultrasonic Method
Quality of Measurement			
Approx. strain sensitivity, $\mu\text{in/in}$	400	300	3 ksi ^(a)
Approx. reproducibility	$\pm 5\text{-}10$ ksi	$\pm 5\text{-}10$ ksi	$\pm 5\text{-}10$ ksi
Determine stress components	Yes	Yes	Possible
Applicable to uneven stress field	Yes	Yes	Possible
Surface or interior strains	Surface	Surface	Interior stress
Effects from Variations in Materials	Affected	Affected	Affected
Field Application			
Field (F) or laboratory (L)	L,F	L	L,F
Remote indicating	No	No	Yes
Cost and Time			
Complexity of equipment ^(b)	4	4	4
Time required ^(c)	3	3	1
Required operator skill ^(d)	4	4	4

(a) Sensitivity in stress is given: $\text{ksi} = 10^3 \text{ psi}$.

(b) 1 to 4: Less expensive $\rightarrow$ expensive.

(c) 1 to 4: Short $\rightarrow$ long.

(d) 1 to 4: Less experienced $\rightarrow$ highly experienced.

The average sensitivity in X-ray residual stress measurement is approximately 5,000 psi (20,22), and the reproducibility is also within this sensitivity at a point in a given specimen. The measurements at the different points in the specimen yield greater uncertainty in reproducibility. However, it has been found that the difference in the residual stress value obtained by X-ray methods and that of destructively measured specimens is generally greater than 5,000 psi.

The advantages of the X-ray diffraction method are:

- (1) It is nondestructive
- (2) By use of the two exposure technique, measurements on the unstressed material can be used to establish a base line
- (3) It measures a very small area of the specimen (0.5 - 0.05 inch).

The disadvantages of the X-ray diffraction method are:

- (1) The grain size of the metal may cause scattering
- (2) Only surface stresses can be measured (maximum of 0.001 inch)
- (3) The surface must be smooth
- (4) Variation in chemical composition effects the test results.

Good accuracy is obtained only with specimens that yield sharp diffraction lines. For example, quenched steels or cold worked metals give diffuse diffraction lines that produce very large errors. Thus, the X-ray diffraction techniques are applicable only to crystalline materials having randomly oriented small grains.

An isotopic source (23) instead of an X-ray source has been used for residual stress measurements. To date the results have not been too encouraging due to low intensity of the source.

ELECTRICAL METHOD

The electrical conductivity of a metal depends on its state of mechanical stress. The elastic strains cause distortion of the crystal lattice and disturb the stability of the atomic siting. This results in an increase in the scattering of electron waves. The change in electrical conductivity can be determined by passing direct or alternating currents directly through the specimen. This effect, sometimes referred to as elastoresistance, can also be measured using eddy current techniques-contactless techniques.

The effect of mechanical stresses on electrical conductivity is given by:

$$\frac{1}{\gamma} = \frac{1}{\gamma_0} (1 + \gamma\sigma) \quad (6)$$

γ = electrical conductivity of stressed material

γ_0 = electrical conductivity of unstressed material

α = stress coefficient

σ = stress

If the stress coefficient is positive, it means that the electrical conductivity of metals is reduced by tensile and increased by compressive strains.

Klimov (17) has measured the stress coefficient and the data given in Table II is taken from his work.

TABLE II
ELECTRICAL CONDUCTIVITY AND STRESS COEFFICIENTS

Material	Electrical conductivity (ohm ⁻¹ m ⁻¹)	Stress coefficient $\alpha \cdot 10^{11}$ (m ² /newton ⁻¹) at frequencies of		
		2 kc	10 kc	50 kc
Austenitic steel	1.4×10^{-2}	1	2	4
Brass	13×10^{-2}	5	14	50
Duralumin	28×10^{-2}	10	25	60
Copper	57×10^{-2}	8	20	35

The values of the stress coefficient obtained are in good agreement with experimental data on the effect of elastic strains upon the electrical conductivity of some metals and alloys (24,25). The view that electrical conductivity of metals and alloys depends upon elastic and plastic strains is also supported by other workers. The dependence is affected by the nature of the metal, its physical and chemical state, and ambient conditions. All other factors being equal, the stress coefficient has been found to be constant for stresses up to $246 \cdot 10^6$ newton/m² (25 kgf/mm²) in the case of the austenitic steel, and up to $98.1-147.0 \cdot 10^6$ newton/m² (10-15 kgf/mm²) in the case of the other materials. As shown by the table, the stress coefficient depends upon the nature of the material, the frequency of current used, and the physicomachanical state of the specimen. With increasing excitation current frequency the induced eddy currents are displaced toward the specimen surface because of the skin effect. The scatter of the values of stress coefficient of specimens of a given material did not exceed 10-15%; the accuracy of measurements was 3-5%.

ULTRASONIC METHOD

The velocity and attenuation of ultrasonic waves are influenced by the physical and mechanical properties of the medium. Mechanical stresses in an

isotropic solid cause a change in the velocity of an ultrasonic wave propagating through it. Stress causes the grains to become elastically anisotropic. This anisotropy affects the velocity of longitudinal, shear and Rayleigh (or surface) waves. The changes are small, caused by the departure from linearity of the stress-strain relationships. Thus, it is possible to detect the changes in ultrasonic velocity or attenuation and relate them to the residual stresses in the specimen.

The velocity of an elastic wave, v , propagating through a homogeneous elastic medium is given by

$$v = \sqrt{\frac{E}{\rho}} \quad (7)$$

where E is the elastic modulus and ρ is the density. A change in E or ρ would affect the velocity. The density and elastic modulus of metal change, in general, as a compressive or tensile stress is applied. The changes in the elastic constants are a second-order effect. This change in ρ is small compared to the change in E and in most cases can be ignored. The fractional change in velocity is very small--approximately 0.0005% per 10^4 psi--in most metals. Only limited applications of ultrasonic techniques have been reported in residual stress analysis (26). Most of the ultrasonic techniques are in the developmental stage, and have been used primarily in laboratory studies.

The techniques developed in recent years for accurately measuring velocity are capable of detecting the velocity difference of one part in 10^6 . In some single crystals and polycrystalline metals, this change corresponds with a stress of less than 10^3 psi (27). The fact that the velocity changes in commercial metals caused by the other factors (such as inhomogeneity, preferred orientation and grain structure) are approximately the same order of magnitude as the changes caused by the residual stresses complicates the stress analysis in many applications. Small changes in temperature can also cause a significant change in velocity. Assuming that the effects due to materials variability are eliminated by the relative measurement of the ultrasonic velocity in stress-free and a stressed specimen of the same material (except for state of stress), the predicted sensitivity of the stress measurement is of the order of 104 psi.

The attenuation of ultrasonic waves in metals, is also a function of the stress in the specimen. From attenuation data, the state of residual stress in a specimen can be determined (28). However, the experimental difficulties in making accurate and repeatable attenuation measurements make this technique less desirable than the velocity technique.

A number of precision experimental arrangements for absolute velocity measurement have been described by Forgoes & McSkimin (26,27). These systems are capable of determining the transit time to a few parts in 10^5 or better, and had been able to determine third-order elastic constants in single crystals. The "sing-around" technique, as it has been called, is very convenient for measuring small differences in the transit time of ultrasonic pulses. Figure 3 shows a schematic diagram of the electronic instrumentation. In the sing-around technique a pulse from the oscillator drives the transducer. An ultrasonic wave is propagated along the specimen for a given distance and the signal detected by a second transducer. The received signal re-triggers the pulsed oscillator for the next pulse. The repetition frequency of the pulsed oscillator is the reciprocal of the electronic delay time and the acoustic travel time. For a fixed distance between the transducers, the wave velocity can be

Fig. 3 Basic Apparatus for Measuring Rayleigh Wave Velocity by the Sing-Around Method.

calculated. Using this modified sing-around technique the ultrasonic wave velocity can be measured accurately to 1 part in 10^5 .

For measuring surface residual stress Rayleigh or surface waves are used (29). This type of wave is propagated over the surface of a solid. Approximately 90% of the energy in the wave is concentrated in a layer on the surface one wavelength in depth. By varying the frequency the effective "depth of penetration" of the wave can be changed.

Of prime importance in the application of Rayleigh waves for measuring surface residual stress is their generation and detection in the medium of interest; this requires adaptation to complex geometries, roughness of the surfaces, different surface shapes, and other characteristics of the surface. The three general techniques of generating surface waves are:

- (1) Use of mode conversion of compressional waves incident upon the surface at a critical angle (29).
- (2) Use of a comb structure to provide spatially periodic coupling of a compression wave to the surface (30).
- (3) Use of specially dimensioned Y-cut quartz crystal in contact with the surface (29).

In the present work, surface waves were generated by a variable-angle transducer system. The angle of incidence of longitudinal waves to a specimen was adjusted such that the following relation was satisfied:

$$\frac{V_L}{V_R} = \sin \theta_{cr} \quad (8)$$

where

V_L = velocity of longitudinal wave in the wedge

V_R = Rayleigh wave velocity generated in specimen

θ_{cr} = critical angle for the Rayleigh wave

Typical results obtained in an experimental study, sponsored by the Air Force, using Rayleigh waves for measuring surface residual stress is briefly reviewed (31). Experimental results obtained on 4340 steel, 7075 aluminum, and a titanium alloy (Ti-6Al-4U) at three different frequencies are given in Figure 4. The results show that for these materials the fractional change in velocity

Fig. 4a Ultrasonic Calibration Curve for 4340 Steel.

of the surface wave is proportional to the applied stress. The slopes of the stress curve vs. velocity change, at different frequencies are the same, within experimental error. These results indicate that the surface residual stress can be determined to $\pm 10,000$ psi.

Using these calibration curves, the stress value determined by ultrasonics was compared with destructively determined stress-depth curves for

Fig. 4b Ultrasonic Calibration Curve for Ti-6Al-4V.

Fig. 4c Ultrasonic Calibration Curve for 7075 Aluminum.

several specimens. A group of ground specimens were prepared under what was assumed were identical conditions, keeping all the grinding parameters constant. Experience indicates this is probably not a valid assumption. Some of these specimens were used to determine the stress pattern by the layer removal technique. The others were used in making the ultrasonic measurements. The correlation between ultrasonic measurements and destructively determined stress-depth curves was poor. The same type of results were obtained with the correlation of X-ray and destructive measurements. There are two major reasons for poor correlation. There is a variation in residual stress along the length of ground specimens, as indicated by destructive tests. The residual stress, determined destructively, gives an integrated value for the entire length of the specimen, the ultrasonic measurement gives an integrated value over a length of 1 to 2 in. (the distance between the transducers) and the X-ray measurements give the stress at a point. The poor correlation between ultrasonic and X-ray measurements and destructively determined stress-depth curves can also be attributed to the problem of reproducibility in the destructively determined stress-depth curves.

A more significant indication of the capability of the ultrasonic technique is to compare the ultrasonic and X-ray residual stress determinations. In this study specimens of 4340 steel were prepared by grinding. In the as-ground condition, X-ray residual stress determinations were made at positions 1/2 inch either side of the center of the specimen, the ultrasonic determinations were made at frequencies of 5, 10 and 15 Mc with the transducers set 1 to 1-1/2 in. apart over the center section of the specimen. After the X-ray and ultrasonic measurements were made, a layer of material was removed from the surface of the specimen by electropolishing. Then, the X-ray and ultrasonic measurements were repeated. The process of layer removal measuring was repeated several times to obtain residual stress data as a function of depth.

These results are given graphically in Figures 5 and 6. The data show that the magnitude of the residual stresses determined at a given frequency decreased as the stressed layers were removed from the surface of the specimens. The rate at which the residual stress decreased as a function of depth was approximately the same for both 15 and 10 Mc surface waves. This is only approximately true for 5 Mc surface waves. For the 5 Mc waves, after a given amount of material has been removed, the values of residual stress remain relatively constant. This variation in the results of 5 Mc can be explained by at 5 Mc the wavelength is approximately 20×10^{-3} in. Thus, as the stressed layers are removed, the interior part of the specimen has more effect on the measurements than does the immediate surface where only small residual stress remains. The residual stress values determined at higher frequencies are greater than those determined at lower frequencies. The values of maximum stress determined by the ultrasonic method were lower for all specimens than were the maximum values obtained by the X-ray method. This was expected, since the X-ray method interrogates the stress in a smaller volume of the specimen than does the ultrasonic method.

The 15 Mc ultrasonic data correlates fairly well with X-ray data. This suggests that if the X-ray value can be used as the reference, the residual stress at various depths can be plotted from the values determined by the ultrasonic technique. The X-ray technique could be used as a primary standard and used to calibrate the ultrasonic technique.

SUMMARY

The nondestructive methods of measuring residual stress are, except for the X-ray technique, still laboratory methods. Experimental work indicates there is some correlation between the X-ray and ultrasonic measurements.

Fig. 5 Ultrasonic and X-ray Measurements of Residual Stress for 4340 Steel Specimen as Functions of Amount of Material Removed.

Fig. 6 Ultrasonic and X-ray Measurements of Residual Stress for 4340 Steel Specimen as Functions of Amount of Material Removed.

Although the layer removal correlation between ultrasonic and X-ray measurements does not represent the actual stress distribution of the specimens tested prior to the removal of each layer, the X-ray value does represent the stress values at the surface after each layer removal. Therefore, the correlation shows the feasibility of ultrasonic determination of surface residual stresses. During the experimental work no attempt was made to determine the effect of surface roughness, impurities in or on the surface of the specimens, or the effect of materials variability. Much additional experimental work is required in order to determine if the ultrasonic technique can be used to give a realistic and meaningful measurement of surface residual stress.

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An Ultrasonic Method for Nondestructively Detecting Radiation-Induced Embrittlement in Pressure-Vessel Steels

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A well documented effect of neutron radiation on carbon and low alloy steels is an increase in the nil-ductility-transition temperature. For certain steels, the transition temperature can be shifted so far upward as to exceed reactor pressure vessel operating temperatures during periods of startup and shutdown. Under some circumstances, operation at or above the transition temperature could be unsafe. Thus, a knowledge of the transition temperature is of vital importance.

A shift in transition temperature is usually measured by conducting impact tests on irradiated Charpy V-notch or other types of impact specimens. Although impact methods are effective, they have a limited sphere of usefulness since considerable quantities of irradiated impact specimens must be on hand to conduct the tests.

Irradiation damage to metals has proved to be difficult to measure or even detect using nondestructive testing methods. However, a recently developed ultrasonic method, the critical angle method, has been shown to be sensitive to physical properties changes that take place as a result of neutron irradiation. Furthermore, a preliminary correlation curve has been generated which shows the experimentally observed relationship between ultrasonic signal amplitude and neutron fluence.

The critical angle method utilizes Rayleigh (surface) waves and thus is sensitive only to surface and subsurface properties. When a sound beam, traveling through water, is

incident on a metal at the Rayleigh critical angle, a lateral displacement will take place along the interface for a short distance before reflection takes place. Figure 1

$$\Delta = \frac{2 \lambda \rho_1}{\pi \rho} \left\{ \left[\frac{a(a-b)}{b(b-1)} \right]^{1/2} \left[\frac{1+6b^2(1-C) - 2b(3-2c)}{b-c} \right] \right\}$$

where Δ = Beam Displacement ρ = Water Density
 λ = Wavelength in Water ρ_1 = Metal Density

$$a = \left(\frac{V_S}{V_W} \right)^2, \quad b = \left(\frac{V_S}{V_R} \right)^2, \quad c = \left(\frac{V_S}{V_L} \right)^2$$

where V_S = Shear Wave Velocity in Metal
 V_R = Rayleigh Wave Velocity in Metal
 V_L = Longitudinal Wave Velocity in Metal
 V_W = Velocity in Water

FIGURE 1 Expression for Lateral Displacement of Ultrasonic Beam at the Rayleigh Critical Angle

shows the factors that will affect the amount of displacement.⁽¹⁾ The critical angle itself is determined by Snell's law, and for steel is about 31°. Figure 2a shows a Schlieren photograph of a sound beam at less than the critical angle; note that the reflection takes place at the points of incidence. Figure 2b is a photograph of the lateral displacement taking place. Since the incident beam contains some beam spread, portions of the beam will be directly reflected; only that portion at the Rayleigh critical angle will experience lateral displacement prior to reflection.

As the incident angle changes, the phase of the reflected beam continuously changes. Figure 3 shows that over a quite narrow range of incident angles around the Rayleigh critical angle, there is a rapid change in phase angle (ϕ) of the reflected beam covering 180° in the space of a few degrees change in the incident angle.⁽²⁾ If a receive crystal is selectively positioned, such as is shown in Figure 4, it will intercept equal amounts of directly reflected energy and laterally shifted energy that have phase differences of 180°. Under these circumstances, an

a

b

FIGURE 2 Schlieren Photographs Showing Beam Shift and Amplitude Redistribution a) no lateral displacement at incident angle less than Rayleigh critical angle b) lateral displacement of portion of beam incident at Rayleigh critical angle

approximate energy balance or null exists. If the material properties were to change in such a manner as to affect its density, attenuation of ultrasound, or acoustic propagation velocities, then the critical angle, and/or the amount of lateral displacement, and/or the amount of directly reflected and laterally displaced energy will be changed. The original energy balance (or null) will no longer exist. Figure 5 shows the appearance of the electrical waveforms. In the

FIGURE 3 Phase Shift of Reflected Ultrasonic Beam Incident on Typical Water-Metal Interface

FIGURE 4 Simplified Representation of Ultrasonic Beams and Crystal Positioning in Obtaining Critical Angle Method Null Balance

Unirradiated

Irradiated $1.8 \times 10^{19} \text{ n/cm}^2 > 1 \text{ MeV}$
 $T_1 = 10^\circ\text{F}$ $T_2 = 300^\circ\text{F}$

Irradiated $1.1 \times 10^{19} \text{ n/cm}^2 > 1 \text{ MeV}$
 $T_1 = -30^\circ\text{F}$ $T_2 = +230^\circ\text{F}$

Irradiated $1.0 \times 10^{20} \text{ n/cm}^2 > 1 \text{ MeV}$
 $T_1 = +10^\circ\text{F}$ $T_2 = +410^\circ\text{F}$

FIGURE 5 Ultrasonic Critical Angle Test Results on Non-irradiated and Irradiated Reactor Pressure Vessel Steel Samples

upper left photograph, the receive crystal is adjusted for an energy balance on a nonirradiated steel specimen. When material properties change--as caused by irradiation--the null balance is disturbed, as shown in the other three waveforms, and a large signal appears where a small one was before. A pulsed mode is used to avoid "standing-waves" and cross coupling interference from the transmit waveform. The signals on the leading and trailing edges appear because of harmonic distortion present on the leading and trailing edges of pulsed oscillations.

To conduct an investigation into critical angle methods response to irradiated steel, it was first necessary to procure irradiated test specimens. Figure 6 lists the neutron

<u>Flux, nv >1 MeV</u>	<u>Fluence, nvt >1 MeV</u>	<u>Irradiation Temp.</u>
1.5 x 10 ¹²	2.7 x 10 ¹⁸	< 350 °F
2.5 x 10 ¹²	4.5 x 10 ¹⁸	
3.5 x 10 ¹²	6.4 x 10 ¹⁸	
3.5 x 10 ¹²	1.3 x 10 ¹⁹	
5.6 x 10 ¹³	1.0 x 10 ²⁰	
1.0 x 10 ¹⁴	1.8 x 10 ²⁰	
1.7 x 10 ¹⁴	3.0 x 10 ²⁰	~ 600 °F
5.5 x 10 ¹²	2.0 x 10 ¹⁹	
2.8 x 10 ¹³	1.0 x 10 ²⁰	
1.7 x 10 ¹⁴	3.0 x 10 ²⁰	

FIGURE 6 Irradiation Program Used for Ultrasonic Detection of Neutron Irradiation Damage in A302-B, A542-II, and 304 SS Steels

exposure given the ten groups of four specimens each. Seven groups were irradiated at low temperatures to neutron fluence levels ranging from 2.7 x 10¹⁸ to 3.0 x 10²⁰ nvt > 1 MeV. Three groups were irradiated at higher temperatures to neutron fluence levels of 2, 10, and 30 x 10¹⁹ nvt > 1 MeV. The purpose of the higher temperature exposure was to determine if the ultrasonic response to the high temperature exposed specimens would be less than that to the low temperature exposed specimens at the same neutron fluence, since a significant amount of in-reactor annealing takes place at temperatures above about 450 °F.(3) Figure 7 shows the test specimens and the manner in which they were encapsulated. Specimens were machined to be similar to a Charpy specimen: cross-section 0.394 x 0.394 in., length 2.577 in., surface finish 16 μ in. Three different materials were selected: A302-B, A542 class

FIGURE 7 Irradiation Program Test Specimens and Capsules

II, and 304 stainless steel (SS). The method of encapsulation was to enclose and weld in a square 304 SS capsule. Wall thickness was about 38 mils, and gap between specimen and capsule varied from 1 to 11 mils depending on desired irradiation temperature. Welding was performed in a helium glove box at one atmosphere pressure. Reactor operating temperature of each group was determined by inclusion of a temperature monitor capsule alongside the test specimens' capsules.

A block diagram of the ultrasonic tester is shown in Figure 8. It consists of a gated oscillator as the transmitter, a gated receiver, detector pulse-stretcher, integrator, and a 2-channel strip chart recorder. The null region in the received signal is gated and converted to a dc level that varies in amplitude as the null region becomes unbalanced. Since normal variations in material properties will affect the null-balance of the critical angle adjustment, it is necessary to scan the length of the specimen and integrate to obtain the average reading. The average reading of several scans on different sides of a specimen are averaged to obtain the final data point representative of the particular specimen being tested.

The mechanical apparatus used in taking measurements is shown in Figure 9, as seen through the window of a hot cell.

FIGURE 8 Block Diagram of Instrumentation Used for Ultrasonic Critical Angle Method Investigation of Neutron Irradiation Damaged Steels

FIGURE 9 Ultrasonic Critical Angle Method Mechanical Apparatus as Seen Through Window of Hot Cell

Figure 10 shows test results on seven A302-B and seven 304 SS test specimens. Measurements were conducted at 2 1/4 MHz throughout this series. These curves would indicate that the property changes due to low temperature irradiation corresponding to as low as 30 to 50 °F increase and

FIGURE 10 Effect of Neutron Irradiation on Ultrasonic Critical Angle Signal Amplitude

as high as over 400 °F increase in the nil ductility transition temperature may be detected using the critical angle method. Only one data point for each metal is available for high temperature exposures. Note, however, for the high exposure of 3×10^{20} nvt, the signal amplitude for the high temperature specimens is considerably less than that for the low temperature specimens. This suggests in-reactor annealing. The high temperature specimens were irradiated at a maximum temperature of 610 °F. For comparison, the integrated readout of the nonirradiated control specimens are shown as the dashed lines. No irradiation damage comparisons should be made from the relative positions of the two tested metals' curves, since data for the two metals were taken under different instrument sensitivity conditions.

Since the irradiated specimens were subjected to a wide range of temperatures during their in-reactor exposure, it was decided to prepare groups of control specimens that had been heat treated for 21 days--the same length of time they were in the reactor--from temperatures of 200 to 800 °F at increments of 100 °F. The results of testing these speci-

mens--except the 700 °F and 800 °F specimens which are still in the furnaces--are seen in Figure 11. There is essentially no change as a result of heat treatment.

Additional irradiated and nonirradiated samples will be examined as they become available, and the data presented herein were only obtained within the past month and, as such, will require additional analysis to substantiate their validity. Nonetheless, it would appear that the critical angle method is capable of detecting property changes caused by neutron irradiation. Because the test readout curve for 304 SS approximates the shape of the curve for A302-B, there is some doubt that a direct correlation to nil ductility

FIGURE 11 Effect of In-Reactor Simulated Heat Treatment on Ultrasonic Critical Angle Signal Amplitude

transition temperature can be made, since A302-B exhibits a typical sharp breaking transition curve whereas 304 SS does not exhibit such a break.(4) Furthermore, it is not known if the critical angle method can identify the extent of irradiation damage from several damage sources simultaneously present in metal. Under circumstances where the effects of neutron irradiation can be isolated, such as in surveillance specimens, the critical angle method may prove to be a powerful tool for assessing irradiation damage.

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Discussion on Papers by Parry, Green, Pedersen-Spanner, McClung, and Hunter

PRESSESKEY (AEC): Mr. Green and Mr. Pedersen both state that triangulation is not a problem. Can someone give a number on the expected accuracy of triangulation in real situations, such as Parry described with transducers on the pipe and the flaw somewhere else in the system? Is there any possibility that both ends of the flaw could be located so that one would know how long the flaw is?

PARRY (PPC): On the SM-1A we used approximately 100 pulses or data points to determine these flaw locales. The statistics were rather bad. But, even with this part of manual triangulation, we were able to isolate the noise source to within a 10-inch-diameter area. With the automatic triangulation mechanism that we are now using, we will be able to sample all 4½ hours of data (several million data points), and we hope to get within an inch or so of the actual source locale.

PRESSESKEY: So that, if propagation or emission is occurring from both ends of the flaw, you would be able to tell me its length?

PARRY: We will be able to define an area that possibly is limited by the length of the flaw—say 2 to 3 inches. We are currently performing a series of tests to define the resolution.

GREEN (AEROJET): I think that there is a little more to the question than that: It is not only can you triangulate but also what are the observable relationships between the area to which you triangulate and the material failure process? If you triangulate to an area that is undergoing microcracking, which nondestructive inspection techniques will not find, what good has it done you? You have just created another disturbance factor in the process.

In terms of ultimate sensitivity, rather than the ability of triangulation to isolate an area, stress-wave sources as small as 10^{-5} square inch can be detected. Furthermore, if one takes these test chambers to complete failure and examines the surface metallogically to determine the origin of fracture, then one can make a correlative study with the defined origins of failure.

PRESSESKEY: I do not think that we are talking about the same problem. You are looking at a chamber that is on a test stand, whereas I am thinking about a reactor which is buried in a lot of concrete shielding and thermal insulation and is thus inaccessible. I do not think that we can carry any of these reactor primary systems to failure in order to look at the characteristics of the crack.

GREEN: I agree, and that is why we have tried to define how much area is swept out by an advancing flaw in terms of amplitude relationships, or in terms of how close one is

to ultimate failure. But that does not necessarily help you because you do not know how big that flaw was to begin with.

MUELLER (BENDIX): I want to comment on a statement just made by Mr. Green. He indicated that the acoustic emission method has a sensitivity unequaled by other nondestructive testing methods. Our work shows that optic correlation techniques seem to have a sensitivity of about the same nature. Characteristically, on fatigue testing we see very significant changes taking place in the first, say, 100 cycles, and then the thing calms down. Finally, the phenomenon that Mr. Green called "pop-in" starts the failure. I wanted to indicate such possibility.

HOLT (AEC): I understand, Mr. Green, that you have been using 2½-inch maximum wall thickness pressure vessels. Now for 10-inch-thick vessels, which is more realistic for reactors, the noise source would be almost a point source, and I would expect a rather large intensity attenuation—a $1/R^2$ attenuation or something of that order. How will this wall thickness affect your method?

GREEN: I cannot give you a direct answer because our experience has been with wall thicknesses up to about 1 inch on actual chambers. But I am not certain that the thickness has any greater effect on attenuation than just a change in total distance from the source. We have had noise sources as far away as 16 feet from the nearest sensor, and still we managed to detect incipient failure.

Sensitivity and calibration can be established in the field at relatively nominal cost by exciting a pressure vessel with an input of, say, 0.1 G at different locations and moving the sensors around the vessel if it is accessible.

MILLER (ORNL): Mr. Parry, you showed a slide of a test pressure vessel having the midsection thickness reduced, and you also displayed the acoustic traces from an embrittled and an annealed vessel. First, did you try to locate the sources of the signal on this vessel? Were they at the machined fillet or at what looked to me like a weld undercut on the slide? Second, what do you mean by "embrittlement" or "annealed," and have you studied the specific treatments (postweld heat-treatment, normalization, quenching and tempering, and hot overstepping) that might be applied to an actual vessel?

PARRY: We had several transducers on each specimen tested; some of these specimens were notched and some were not. In all cases we did locate the source of emissions. In the unnotched specimens the signal was coming from the reduced, high-stress region in the center of the jugs. In the notched specimens the signal was coming from the notch.

Regarding embrittlement and annealing, the complete jug was heat-treated. We tried to achieve two extreme conditions: highly embrittled and highly annealed. In the near future we plan to examine different degrees of annealing or different amounts of quenching.

GREEN: Mr. Hunter, if I understand you correctly, the technique you are using is related to the autoelasticity technique of Carlson from some time back. That technique was originally used to study residual stresses in the material. Are you suspending your samples so as to preclude the establishment of residual stresses as a material change property while you are testing?

HUNTER (BATTELLE-NORTHWEST): I am not familiar with the Carlson method. It is my understanding that the method we are using is unique to our operations at Hanford. However, the Rayleigh critical-angle method is known to be sensitive to fatigue that would occur in the specimen, and this effect would have to be isolated from the effects of irradiation, corrosion, or subsurface flaws.

The specimens were enclosed in stainless-steel capsules and freely suspended in the spider holders, which were then put in the reactor for irradiation. There was no mechanical working of the specimens in the reactor.

BULLOCK (ORNL): Mr. Pedersen indicated that the frequency of spikes seemed to be directly correlated with each grain-boundary failure. My question relates to the amount

of energy that appears to be observed in these spikes, and whether or not this can be supported by theoretical prediction? It seems overwhelming to me that one grain-boundary fracture would release a detectable amount of energy.

PEDERSEN (BATTELLE-NORTHWEST): We use sensitive transducers and very high gain amplifiers, so that very minute amounts of energy can activate the transducers. We have not presently correlated the energy released with each grain cleavage with the amount of energy which was detected by the transducer. I indicated that the work was preliminary, and it has not been completed.

DUNEGAN (LRL): I would like to comment further on Mr. Bullock's question. I think it is presumptuous to say that the emissions are due to grain-boundary cleavage. Schofield's early work related to slip-band density indicated that in single crystals one could obtain resolutions down to about 50 dislocations. At Lawrence Radiation Laboratory we are currently trying to relate the dislocation density to the acoustic emissions in silicon-iron, and we find that what it being heard amounts to the breakdown of 40 to 50 dislocations, and this is quite small. But acoustic emissions are also obtained from 20 dislocations, from the formation of microcracks. Emission is obtained from crack extension. So the continuous, or high-frequency low-amplitude, emissions can be related directly to dislocation mobility studies.

PARRY: I too would like to comment on sensitivity. We found it to be dependent on the wavelength one is working with. These emissions have a spectrum that goes up into the megacycle range. In our work at 100 kc, we have gone out as far as 150 feet on primary piping and the channels were still saturated with simulated acoustic signals. Thus the attenuation is very small when one is dealing in the low frequencies.

One final comment on sensitivity. Flaw detection capabilities depend on the particular system being used, and I think that we, today, are actually talking about two different areas of detection level. It appears that the stress-wave technique has the ability to look at microflaws, which in some cases might not be necessary. Thus it is a matter of establishing where we actually need to set the threshold of our system in order to detect a flaw that is a meaningful flaw, a flaw that will cause premature failure of a vessel.

GRIFFIN (AI): Mr. Green, you suggested that the SWAT system could be used for quality-assurance or quality-control evaluation or acceptance testing. In a new system, to what level of pressure would you have to go to assure yourself that you had an acceptable system with your live-source type of video observation? Supposing you could see nothing, how would you go about this?

GREEN: This is a philosophical question, and I do not know if I can give you a distinct answer. I can only give two extremes. First of all, we are dealing with a basic premise that any manufactured structure consists of defects and flaws. Some of these may be totally innocuous; others may cause catastrophic failure at less than what you have labeled 100% of the design life.

HAKE (AECL): If, instead of fast crack propagation being the great worry, creep were the limiting factor, could this sound emission technique be used to give some idea of how we are approaching our limiting conditions?

GREEN: Yes.

BULLOCK: Mr. McClung, you did not mention neutron radiography in your paper. Would you comment on the state of the art?

McCLUNG (ORNL): I do not think we have time to discuss the state of the art of neutron radiography. Some people have suggested the use of cold neutrons for evaluation of pressure vessels because of the particularly inviting attenuation characteristics of cold neutrons in steels.

TENNEY (LASL): I would like to add one final philosophical comment. We have just reached the halfway mark of the conference, and therefore it might be appropriate to say a few words in the spirit of constructive criticism. I wish to refer to the keynote address that Mr. F. Schroeder, Jr., gave yesterday morning, in which he defined the purpose of

The Role of Nondestructive Testing in Industry.

the conference. He stated that in the commercial reactor industry a great need exists for high-quality production techniques, high-quality inspection techniques, and methods for detection of minor failures in the incipient stage. Unfortunately, economic and safety considerations often seem to be antagonistic.

The subjects that have been presented thus far were indeed of great scientific and technical interest. On the other hand, if we are to follow the theme suggested by Mr. Schroeder, then some consideration should be given to his request. The nuclear industry is growing at a very fast rate (see the September 1967 issue of *Nuclear News*), and, as a result, the need for having sound guidelines for safe reactor construction has become more urgent. How does nondestructive testing (NDT) fit into this situation? I believe that at present we have two extreme situations as far as quality assurance through nondestructive testing is concerned:

1. Close coordination and dissemination of knowledge in the field of nondestructive testing as applied to fuel elements has been in existence for at least 16 to 18 years. The national laboratories have developed sophisticated NDT equipment, most of which had to be built by the laboratories themselves. The knowledge is well-documented and is available to all interested parties. On the other hand, the commercial NDT equipment industry at large has not been fully interested in building these sophisticated instruments because it has not been clear to them to what extent such an approach would lead to mass production. The problem of inspecting fuel elements on a production-type basis by private industry has to face two major bundles: lack of commercially available equipment and lack of skilled manpower.

2. The opposite situation prevails in the mass production of pressure vessels, pumps, valves, tubings, welds, and related equipment. Those industries which have been active in the manufacture of such parts on a conventional basis do not realize the stricter approach necessary to guarantee the quality of those parts when used in nuclear reactors.

During the last few years, it has become more and more apparent that the role of non-destructive testing in industry is an ever-increasing one. The accompanying figure establishes the position of nondestructive testing as a pivotal point for the entire manufacturing process. The continuous feedback with research and development, materials engineering, process engineering, and design engineering is a must in order to ensure high quality.

I wish to close by saying that it is of paramount importance to establish a high-level coordinating committee on testing in the reactor industry in order not to duplicate various approaches but to save manpower by joining forces.

**Methodology for Building into Nuclear Power
Plants Provisions for On-Line
Testing and Inspection**

R. B. Briggs, Chairman

Introductory Remarks

R. B. BRIGGS
Oak Ridge National Laboratory

I would like to try to set the stage for the papers that follow. Whether testing is required in a reactor plant is not a matter of question. The question is how much testing? The operator wants it. The insurance company wants it. And the regulatory authorities require it, as was pointed out in yesterday's talk.

I think that most kinds of testing can be accomplished easiest and at least expense with the reactor plant shut down. However, the important thing in a power industry is to keep the reactor plant running, and I think that one can see that power plants will run from 1 to 5 years, certainly, in the future without requiring shutdown for repair of mechanical components or reactor vessels and without requiring shutdown for change of the machinery that is going to do the on-line refueling. So availability becomes a very important consideration. And then a major consideration in the design becomes how much testing shall we do? How much shall we pay for it? What method shall we use?

I am not really very familiar with this term "methodology," but I assume that the talks this afternoon will explore these facets: What is the expense of including on-line testing? What methods can be used? How do we do this in the design and in the construction of the plant? What provisions do we make?

The Benefits and Limitations of Designing for Nuclear Power-Plant Safety-System Testability

N. K. SOWARDS
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As the title of the conference implies, there are two rather different viewpoints from which one can address this basic problem of plant and system testability. One of these is with respect to the plant safety and what is best from the safety point of view. Certainly another valid perspective is from the standpoint of the plant availability considerations. Being in the nuclear safety business I have elected to slant my remarks toward the safety part of the problem; however, the two are closely related in a number of ways.

As Frank pointed out in the keynote address, we in the nuclear safety industry face a rather unique problem in that the potential magnitude of accident consequences places us in the position of having to take all reasonable precautions to attain absolute assurance of safe plant operations.

A situation analogous to this may exist with respect to nuclear weapons systems; however, in this case the device in question is more specifically limited with respect to interdependence effects, and the organizational and administrative responsibilities do not cross the boundaries of private industry and government to the same extent.

Certainly the most direct approach to achieving the "safe nuclear plant" is the implementation of ultra-high engineering standards of design, fabrication, installation, and inspection. This will take us a long way toward preventing the accident from occurring in the first place. The overall safety of the plant is further enhanced by the application of numerous safeguards systems directed toward limited accident consequences. It is principally these consequence limiting safeguards toward which my remarks are directed. However, these safety systems can be viewed as additional accident barriers only if there is a high degree of assurance that they will operate as designed when they are called upon to do so. For systems that do not play a role in the normal plant operation and function, but which sit idle for months or even years, very special design measures and procedures must be applied if we expect to achieve this high degree of assurance. Periodic operability and performance tests of these systems add greatly to our assurance that they will do the job for which they are designed, if and when they are ever needed. To permit the performance of effective operability tests, I think we must recognize that the testing requirements must be a partner and contributor to establishing the system design requirements. We must define the in-plant tests to be periodically performed and then develop the necessary capability within the system design for performing these tests without compromising the primary function of either the plant or the specific system under consideration. As an example of the degree to which the testability requirement can effect the system design, let's consider the ECCS concept shown in Fig. 1.

In this simplified example, the emergency coolant pump operates continuously developing a total discharge head somewhat below primary loop pressure. Sufficient flow is bypassed through valve B to prevent overheating of the pumps. In the event of a loss-of-coolant and a large drop in primary loop pressure, the check valve or rupture disk will open allowing emergency coolant to flow to the core region. Obviously there are practicability and feasibility considerations such as the ability to design these very large pumps to operate at essentially no-flow

Fig. 1 A continuously operating Emergency Core Cooling System for power reactor.

conditions without developing problems such as cavitation damage and excessive erosion of the rotor. The example is presented simply to show the extent to which the testability requirement can and should be reflected in the system design and operation.

In particular, if we are to ever achieve metropolitan siting, there must be absolutely no question as to the availability and effectiveness of emergency core cooling in the event of a loss-of-coolant accident.

Of course, the major benefit of designing systems for testability is to improve the level of assurance of the system operating when needed; however, a more specific benefit of designing safety systems for testability is that if properly used it can enhance the overall plant availability. This is related specifically to the amount of plant shutdown time required to periodically test the safety systems. Systems that have been specifically designed with the test requirements in mind can normally be tested quickly and effectively with a minimum of perturbation to the plant operation.

As we all recognize, there are major problems and limitations which we must face in designing safety systems for in-plant testability. I think the major limitation is the lack of ability to test under a realistic accident environment. It is of only limited significance to periodically demonstrate, for example, the operability of an electro-mechanical sensing and control system in a calm, collected, moisture free, ambient temperature, shock and vibration free environment when the one important time for the system to operate is a radically different environment including high temperature-high pressure steam accompanied by shock, vibration, and high radiation levels including possible confusion and uncertainty on the part of the operator. This limitation must be adequately examined and the proper component environmental proof tests performed on newly manufactured as well as used and aged components if we are to achieve a high level of assurance of satisfactory system operation in the event of an accident. This is probably the major stumbling block to realistic in-plant testing that is facing us today.

Another limitation which we face in designing safety systems for testability is that related to system complexity. The age old maxim that the simplest system is the best system has important meaning for us, particularly when it is with reference to system reliability. Designing systems for testability can add greatly, if not unduly, to the system complexity. It will most certainly result in additional components, connections, controls, and modes of operation which do not necessarily enhance the ability of the system to perform its basic function. It can

aggravate the "Human Factor" problems by increasing the probability of operating errors such as leaving the system at least partially in the test configuration. In light of this possible compromise of the basic system function one must ask himself the question, "Has he really improved the situation?" "Have we reached a high level of ASSURANCE of the system operating when needed?"

Another limitation we encounter is the frequent situation in which operation of the system is not compatible with normal plant requirements. For example, the only really complete test of a containment spray system is to dynamically activate the system and monitor flow rate and distribution through each and all of the nozzles; however, the undesirability of performing such a test in a modern nuclear plant goes without saying. Still another limitation to testing safety systems is that the interdependence or interrelationship between the systems cannot be accurately represented. This is a result of the fact that the plant cannot be placed in an actual accident situation. Such things as electrical bus loading of emergency power supplies may not be represented with fidelity. Related systems cannot always be placed in the appropriate operating mode to assure that the necessary interrelationships are being mocked up in the test. Certainly it must be left to the ingenuity of the designer to first recognize and secondly to achieve solutions to the many in-plant testability problems such as I have just mentioned.

Although I had not originally intended to discuss the general topic of a methodology for building testability into power plants; after looking over the agenda and seeing the rather specific nature of the papers to be presented, I thought it might be appropriate to mention a few thoughts relative to the general subject in hand.

Methodology is defined as the "Science of Method", which implies a logical scientific procedure for achieving the desired objectives. The key to a methodology for designing testability into systems is a very thorough system engineering approach wherein one defines the requirements or criteria for the system and the tests one wishes to perform to assure the installed system meets these requirements. Then

through an iterative process, selects those system design features which represent the best possible compromise between the many requirements for the system and the ability to test the system throughout the plant lifetime. This can be defined in certain logical steps which are shown in a simplified way.

First of all a clear and precise definition of the system requirements must be developed and a system design established to satisfy these requirements (see Fig. 2).

Fig. 2 Methodology for developing in-plant tests.

In defining the system requirements it is essential that one view in a logical manner such questions as:

What function must the system perform?, or in other words, what do we want the system to do?

What plant operating conditions prevail during its use?

What environmental conditions prevail during its use?

Upon what supporting systems is the operation of the system directly dependent?

What interdependence relationships exist with other systems?

After a system design has been developed fulfilling the design requirements, a group of "Ideal" in-plant tests can be defined to assure the system requirements are met. At this point in the process, this can and should be done without the constraints of the effects on the plant and related systems but rather, "What are the best possible tests to be performed on the system to assure the operability, performance, and effectiveness of the system under test?"

After having developed this group of "Ideal" tests we are then in a position to identify the conflicts between the "Ideal" test requirements and the system-plant limitations. At this point, in the process, an engineering compromise begins. We may be able to eliminate some of the conflicts by modifying the system design; however, after modifying the design it is essential to review the "Modified" system to assure that the system requirements are still met. The "Ideal" tests can then be modified to assure their compatibility with the modified system. After the tests have been modified, we must then go back and identify those areas where the tests are no longer fully verifying that the system is meeting the requirements specified. We are then in a position to design supplementary testing such as component and separate effects tests to compliment the modified "In-Plant" tests and thus assure a more complete verification of system operability and performance.

In summary I think the point is that performing periodically a series of operability tests, no matter how complex, will not necessarily achieve for us a high degree of assurance of safety system function. The situations and conditions unique to the real accident environment must be kept uppermost in the mind of the designer as he develops the system and the provisions for testing it. The designer may find this sort of systems engineering approach a useful tool for achieving a system that is properly optimized not only to meet the system requirements, but to accommodate the performance of tests so as to better assure it will function as intended in the event it is called upon to do so.

Malfunction Detection Systems for Liquid-Metal Fast Breeder Reactors

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INTRODUCTION

For many years computers, both digital and analog, have been an important tool in reactor design and analysis. However, in the past, computers have only been used in a limited fashion in the on-line analysis of reactor operation. It is only recently that the use of on-line computer systems has received more serious consideration. In the thermal reactor field, the new plants being installed during the next few years will have on-line computing systems which will perform data logging of raw signals and a modest amount of analysis. Primary objectives of the thermal reactor application are to carry out core performance calculations and to minimize reactor downtime, with little effort being devoted to the use of the computer as an on-line safety analysis device. With the present trend towards greater core instrumentation, malfunction detection can be made more positively and the malfunction detector aspects can be expected to become more important.

The use of on-line computers is less extensive in the fast reactor field than in the thermal reactor field. Rapsodie does have an on-line computer, but it is used only for data logging and gross performance monitoring. EBR-II makes intermittent use of a computer for on-line acquisition and analysis of data from transfer function measurement experiments. At APDA, there is currently a program for the development of an on-line computer system for Fermi, with the prime goal being the development of the instrument as an on-line safety analysis device. In this capacity the computer would function as a generalized malfunction detection analyzer.

It is anticipated that the use of malfunction detector analyzers on fast reactors will make a substantial improvement in the operational reliability of these plants as well as an improvement in the safety of these reactors. These improvements can be realized from three different sources:

1. A more refined interpretation of reactor output signals.
2. A more complete evaluation of available reactor output signals.
3. An improved dependability of detecting a malfunction by decreasing the reliance placed on reactor operators.

The improved refinement in interpretation is possible because of the capability of the computer to make analytic manipulations of reactor sensor signals rapidly. Thus, it is possible to make complex calculations, to make cross checks on independent signals, to make allowance for change in operating mode, and to check actual against expected output under transitory conditions such as startup or power level adjustment. These analyses can all be done in the real-time mode in order to make the interpretation of value to the operator.

A more complete evaluation of the reactor conditions is also possible by virtue of the capability of the computer to perform calculations rapidly. Signals and interpretations that are not normally processed on a continuous or frequent basis can be so treated with a fast on-line computer. Examples of this type of signal interpretation and evaluation are (1) lubricant levels in primary system component reservoirs, (2) component performance data such as pumping power versus primary system flow, which might be an indication of change in core pressure drop characteristics, (3) thermal efficiency of intermediate heat exchangers, and (4) the performance of steam generators. In most of these applications, the signal interpretation and evaluation is commonly done only on a very infrequent basis. However, the lack of routine frequent checks is primarily due to the complexity of the signal interpretation and the large amount of stored data that must be used in the calculations. This situation is easily remedied with a modest computer.

In addition, an improvement in dependability can be realized through a reduced dependence on the human element. This is a very important item for several reasons. First, the number of output signals from a large power reactor places a strain on the human capacity to monitor all of them with any degree of thoroughness. Second, there are periods when difficulties arise in one part of the reactor system which demand the operator's attention and distract his visual attention from the remaining control panel displays. Third, a reactor will operate in a normal fashion for the large majority of time, and this will have some effect on the alertness of the operating staff.

CONCEPT OF MALFUNCTION DETECTOR ANALYZER

There are two possible basic computer systems that would accomplish the desired objective: (1) analog computers and (2) digital computers. The digital approach is preferred by APDA because of its much greater flexibility. The use of a stored program permits the calculational method to be altered at will, and it permits new calculations to be added to the program as situations dictate. A schematic of a basic digital malfunction detector analyzer is shown in Figure 1. The basic system consists of the reactor system sensors, a signal conditioning and conversion unit, a digital computer, and output display and action-taking devices. Signals from the reactor are transmitted to a multiplexer which switches signals according to computer instructions, conditions the signal, and feeds it into an analog to digital converter. The signals are fed from the analog to digital converter into the digital computer which then processes this information successively through analysis, comparison, and decision-making steps. The decision-making step then initiates the action to be taken. This action could be of three types:

1. Do nothing if the condition is normal.
2. Alarm and display if the condition is abnormal but limited.
3. Scram if the condition is outside tolerable limits.

Fig. 1 - Generalized Malfunction Detection Analyzer

In addition, the computer should have a priority-interrupt feature, which would permit the reactor operator to interrupt the normal cycle of calculation so that he might request a visual display of information or a specific signal interpretation.

The system should also include provisions for display and permanent recording of basic and interpreted signals by means of typewriters, line printers, oscilloscopes, magnetic tapes or punched cards.

The output actions commanded by the computer should be dependent upon the interpretation placed upon the output signals. Each possible malfunction and its output signal would be analyzed a priori and then classified on a priority basis in which the possible consequences of the malfunction and the time available to take action are taken into consideration. One possible classification scheme is shown in Table I. Three classes of signal interpretation are used, and examples of the items that might be included in each are indicated. Definitions of the three classes are as follows:

- Class A - The implications of this type of abnormality are serious and prompt action is necessary. A scram is required.
- Class B - The implications of these malfunctions are serious but the time scale of events is measured in minutes rather than seconds. An alarm is sufficient for initial action. There is time available for the operator to evaluate the output signal and to take corrective action.

Class C - Any departures from normal conditions for this type of malfunction are more likely to take place over periods of hours or days. This departure would be noted in a printed log. Further immediate action is not necessary.

MALFUNCTION DETECTOR ANALYZER PROPOSED FOR FERMI

The future fast reactors will be able to benefit from instrumentation development and more complete instrumentation of the reactor. In the case of an immediate application for a reactor such as Fermi, it is necessary to use existing installed instrumentation. There is also good reason to approach the whole concept of malfunction detection analyzers in simple steps and allow the system to expand as experience and confidence in its working ability grow. A planned system should be designed with later expansion in mind.

For Fermi, the initial malfunction analyzer as proposed (see Figure 2) will treat three items, i.e., reactivity balance, core subassembly ΔT 's, and the cover gas fission product activity (References 1, 2). The recommended digital computer will have a capacity in excess of the requirements for the treatment of these three items and will allow the inclusion of other signals and interpretations when and if desired. Some of the details of the three signal interpretations are described below.

TABLE I Reactor Output Signal Interpretation

<u>Class A</u>	<u>Class B</u>
Core Channel Temperature Rise	Sodium Plugging Temperatures
Reactivity Balance	Sodium Impurity Monitoring
Delayed Neutron Detector	Pump Performance Characteristics
Sodium Boiling	Fission Product Activity in Cover Gas
Change in Lubricant Volumes	Leak Detector
 <u>Class C</u>	
Intermediate Heat Exchanger Heat Balance	
Steam Generator Heat Balance	
Reactor Vessel Temperature and Stress	

The reactivity balance check is an exceptionally sensitive method of detecting an abnormality. Fast reactor power coefficients have been demonstrated to be accurately repeatable as long as conditions remain unchanged. Furthermore, the power coefficient is very sensitive to any change in flow, even localized. This fact has been well demonstrated in the Fermi incident

Fig. 2 - APDA Malfunction Detection Analyzer (Proposed)

of October 5, 1966. In this incident, blockage of about 2 per cent of the core caused an increase in the negative feedback by more than a factor of 3. This observation is discussed in some detail in a later paper of this conference (Reference 3).

1. Reactivity Balance

The computer will compare the reactivity inserted by the control rods against the reactivity adjustment that should have been made to compensate for power override, for drift or fluctuations in inlet coolant temperature, for burnup effects, and for the reactivity due to the neutron kinetics (see Table II). The power override reactivity is obtained from the ion chamber currents, the flowmeter readings, and a built-in knowledge of the power coefficient including its flow dependence. The net effect of these components will be checked against the reactivity inserted by the rods as determined from rod position indicators and the measured calibration curve. Lack of a check within accuracy limits is an indication of a malfunction.

It is estimated that this balance can be performed to an accuracy of about ± 1 cent on the basis of ex post facto calculations made using the data available from reactor operation at Fermi on two separate occasions.

2. Core Channel ΔT 's

In the Fermi reactor, only about 25 per cent of the core sub-assemblies have thermocouples, so the interpretation of the thermocouple data does not ensure malfunction detection in all cases of local blockage. However, despite this limitation, it seems preferable to accept the partial protection rather than none. Also, since this equipment is regarded as a generally useful tool for future designs, it is worthwhile to gain experience with this type of malfunction detection, even in a system with limited instrumentation.

TABLE II Reactivity Balance

Analysis

$$\Delta k_R = \Delta k_{P.O.} + \Delta k_{T_{in}} + \Delta k_{B.U.} + \Delta k_{N.K.}$$

Reactor Signals

Rod Positions
Ion Chamber Current
Sodium Inlet Temperature
Primary System Flow

Frequency

Checked every 30 seconds.

Accuracy

Approximately ± 1 cent.

Notation

Δk_R = Reactivity inserted by control rods.
 $\Delta k_{P.O.}$ = Power override reactivity component.
 $\Delta k_{T_{in}}$ = Reactivity due to coolant inlet temperature change.
 $\Delta k_{B.U.}$ = Burnup reactivity effect.
 $\Delta k_{N.K.}$ = Neutron kinetics reactivity effect.

Detection would have occurred at 8 Mw on October 5, 1966.

The analysis consists of calculating the ratio of the measured channel ΔT to the expected ΔT for each instrumented position (see Table III). The check is made every 60 seconds; the ratio must be unity, with an allowable deviation to ± 4 per cent in order to have assurance of the absence of a mal-

TABLE III Core Subassembly ΔT

Analysis

$$R = \frac{\Delta T_{\text{Meas.}}}{\Delta T_{\text{Pred.}}} = \frac{T_j - T_{\text{in}}}{C_j \frac{I}{F}}$$

Reactor Signals

- T_j = Outlet temperature of channel j.
- T_{in} = Coolant inlet temperature.
- I = Ion chamber current.
- F = Flowmeter reading.
- C_j = Normalization constant for channel j.

Frequency

Checked once every 60 seconds.

Accuracy

$$R = 1 \pm .04 \text{ at } 110 \text{ Mw.}$$

Detection would have occurred at 5 Mw on October 5, 1966.

function. The allowable deviation has been based on demonstrated instrumentation performance.

3. Fission Product Detection

The fission product detector system at Fermi measures the fission product activity of the cover gas system. It is a slow system with a response-time of approximately 13 minutes. Therefore, it has limited value for safety purposes. However, it does have good sensitivity and can detect failure of a fraction of a fuel pin.

The calibration of this system is not yet completed and details of the analysis are yet to be established.

CONCLUSIONS

It is felt that a malfunction detection analyzer, as described above, will make a significant improvement in the safety and operational reliability of the future liquid metal fast breeder reactors. However, it is recommended that these systems be put into use on a gradual basis. This concept of gradual introduction would apply in two ways: (1) the number of malfunctions treated, and (2) the authority given to the system. In the initial stages of use, the authority should be restricted to alarm and operator notification. This limitation should continue until enough confidence in the instrumentation and the malfunction detector analyzer is proven. Needless to say, improvement in instrumentation reliability should be made in conjunction with the development of the malfunction detection systems.

The question of instrument reliability raises another very important question: What are the steps to be taken if a malfunction is indicated? False indications could result in a prohibitively large amount of downtime. Eventually the malfunction detector analyzers should be developed to the extent that several cross checks on an indicated abnormality should be possible. These cross checks should eventually be built into an internal automatic verification process.

For the present generation of reactors, the instrumentation limitations make cross checking impossible in many cases. The Fermi October 5 incident is offered as an example. Assume that the proposed malfunction detector analyzer was in operation and that the blockage did not occur in a subassembly next to a thermocoupled position. At 8 Mwt, and prior to melting, thermal distortion has caused a detectable reactivity discrepancy. However, there is, in this example, no confirming malfunction evidence, nor is there an indication of the location of the problem area. Furthermore, since the observation that something is wrong occurred before melting took place, there may be no identifiable permanent evidence within the affected subassemblies. With only the reactivity anomaly available, it would be an extremely difficult task to deduce the cause of the malfunction.

Obviously a practical approach to follow after malfunction detection has to be developed. One significant step being taken at APDA is the development of a flow-measuring device which can be located over each channel. This device is only usable during shutdown conditions. One possible approach is to make it a matter of policy to shut down following a malfunction indication, hold the same flow conditions, and check each subassembly position for flow if no obvious fault is indicated. There are also other possibilities available at Fermi. These include offset handling machine maneuvers which make it possible to check subassembly handling head positions and lifting forces. These operations were actually very useful in deducing the region of the core in which failure occurred.

The general problem of what steps to take after a malfunction indication is a more difficult one for existing reactors than for the LMFBR demonstration plants and commercial plants to follow. It is anticipated that these plants will be more completely instrumented with core subassembly thermocouples and possibly individual channel flowmeters. With this more complete information available, the potential to rule out false signals quickly by means of cross checking independent signals, or to determine the actual source of a true malfunction will be greatly improved.

ACKNOWLEDGEMENTS

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Designing for In-Service Inspection

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INTRODUCTION

The demands for structural integrity of nuclear reactor equipment over the projected life of the plant has created increased interest in the development of inspection techniques for assuring this integrity. At the present state-of-the-art there are central station power reactor plants in four different categories with regard to inspectability. These are

1. Plants already in service
2. Plants under construction
3. Plants being designed
4. Plants committed but not designed

Since the inspectability of a plant has only recently become a basic design objective, the first two categories are essentially similar and have varying degrees of inspectability depending on the initial plant design and construction details. Generally, it is difficult to make changes in these plants to improve inspection access.

In plants in the latter two categories, changes can be made consistent with safety, economics, and schedular problems so long as the designer has some knowledge of the inspection procedures which are likely to be used. It is at this point where designers must make some predictions as to the nature of inspection equipment which will ultimately be available in order to provide the desired access. In both cases, the nondestructive evaluation engineers must have constant dialogue with designers and operators to assure availability of equipment.

The objective of this talk is to review some of the problems associated with the use of mechanized ultrasonic systems as periodic inspection devices for primary pressure boundaries of nuclear systems, particularly with regard to accessibility for inspection.

REVIEW OF ACRS REQUIREMENTS

Recent criteria released by the Advisory Committee on Reactor Safeguards for new plants have included specific recommendations as to inspectability. They recommend that the inside surfaces of the reactor vessel be available for visual examination and that provision be made for inspection of 100% of the volume of the pressure vessel wall by ultrasonic methods. Reaction to this has been varied depending on the specific manufacturer, however, consideration is being given in each case to improved inspectability, particularly in new designs. On units in operation or under construction each individual unit is generally being investigated to determine the type of inspection possible with the specific design limitations. The basic approach is to monitor the points of maximum stress concentration for a statistical review of the most probable points for defect initiation and growth.

COMPONENT PRESSURE BOUNDARY DESIGN LIMITATIONS

The design limitations to in-service inspection basically fall into two categories - those limitations which are the direct result of geometrical restrictions and those that are the result of material restrictions. In this paper, the materials selection is considered to be part of the design of the components. The limitations which are imposed by geometrical design depend, of course, to a great extent on the nature of the ultrasonic systems which can be developed and the access available for those systems in various locations where inspections are desired. Since for the most part, the areas of greatest interest, as far as service investigation is concerned, are areas of geometrical stress concentration, it is clear there will be some type of geometrical characteristic at these inspection points which must be overcome by the design of the specific inspection equipment. The list in Table I indicates the most difficult geometrical design problems. It is in these areas that we must address our serious consideration as far as in-service inspection is concerned. It is, of course, desirable to have access from both surfaces to inspect these locations although for the most part the inspection studies made so far have been limited to inside surface availability only. The design limitations shown in Table II are those which are considered to be materials related. Some of these are inherent limitations and others may, on further research, be resolved to less serious problems.

One of the major problems that will face the investigators with any component which has not been subjected to previous ultrasonic inspection will be the necessity for analyzing any defects found by the ultrasonic test in comparison with those defects which are acceptable under Code construction.

TABLE I
GEOMETRICAL DESIGN LIMITATIONS

1.	Nozzle to shell junction
2.	Core and vessel supports
3.	Cladding surface finish
4.	Welded in thermal sleeves
5.	Partial penetration welds
6.	Wall thickness of components
7.	Weld groove geometry

TABLE II
MATERIAL LIMITATIONS

1.	Ultrasonic attenuation in cladding
2.	Scattering of ultrasonic waves at the cladding, steel surface
3.	Plate material integrity (lamination) or segregation
4.	Ultrasonic attenuation of austenitic stainless steel and inconel castings and weldments
5.	Possible changes in ultrasonic attenuation as a function of irradiation

Up to this time the normal inspection has been radiographic inspection techniques. Ultimately, vessels, piping, etc., will no doubt be inspected by the same technique which is to be used in the field to provide more or less a fingerprint of the initial condition of the vessel. However, units presently in service or under construction are not going to have such a preservice fingerprinting. Therefore, in the immediate future, there may be some difficulty resolving indications which are found. This may, however, be more of a fear

than a reality in that the inspections made so far by Southwest Research Institute have indicated relatively good correlation between the results of the radiographic inspection and that of the ultrasonic inspection techniques. It has been, however, necessary on occasion to go back to the shop fabrication records to review radiographic and repair sequences.

DESIGN LIMITATION OF INTERNAL STRUCTURES

As a design objective, it is desirable that all internal structures placed inside components be designed so that they can be removed to clear the inside of the vessel for complete inspection. A requirement to remove them is generally not looked upon with favor by the utilities as a routine inspection requirement because of the long time required in the removal of all core structures, thermal shields, surveillance samples, etc. As a result, estimates have been made of periods of two weeks to three or four months for such total removal of internals. Therefore, internals must be examined both from the standpoint of removability as well as inspection access to certain critical areas during the progress of the operation of the unit. Sometimes, access can be provided to nozzles and to certain of the major weld seams by judicious placing of holes through which equipment can be entered. Such a system sometimes becomes quite complicated as in the case of the system proposed for the inspection of the inlet and outlet forced-circulation nozzles for the LaCrosse Boiling Water Reactor. Figures 1, 2, and 3 show a proposed inspection system for the forced-circulation nozzle to shell junction. In this case, the holes in the core support plate which are available by removal of the riser tubes are not lined up with the nozzle; hence, in order to make an inspection from the inside, the mechanism must go down through this hole in the core support plate and make a series of turns to be positioned in the nozzle. What would normally be a relatively easy engineering job if the holes had been lined up with the nozzles becomes a very difficult and expensive project. Unfortunately, this might be the only way to provide a reasonable inspection of this area.

One of the important features of internals is the ability to inspect the internal systems themselves. Of the components which have failed in operating reactors to date, many of the more serious ones have occurred in internal systems. Increasing concern should be given to providing inspectability of the internals themselves and particularly those which affect reactor safety such as control rods, supplemental core cooling systems, etc.

Another consideration which must be given is that components may not be completely dimensionally stable. Initial clearances designed into a unit must of necessity be very small and shifting may occur in service. If such internals freeze inside, it will be necessary to take alternate approaches to in-service inspection.

Figure 1. Plan View of Inspection Apparatus and Its Relative Position with Respect to Access Hole in Core Support Plant

All personnel connected with plant maintenance and construction should be familiar with inspection access provisions. The natural desire to prevent components from breaking loose in the reactor generally dictates strong tack welds on bolts, etc., so that sometimes structures designed to be removed cannot be because of over zealous staking procedure.

INSULATION AND BIOLOGICAL SHIELDING LIMITATIONS

To cover the overall primary system of a reactor plant, it is necessary to discuss also the influence of a variety of external structures which may reduce access to certain critical parts of the plant. For example, insulation is applied by a number of different procedures, depending on the particular design group designing the plant. Some of these can be removed very readily from certain areas and others are very difficult to remove and, in some reactor plants, it is nearly

Figure 2. Vertical Cross Section Through the Junction of the Forced-Circulation Nozzle and Shell

impossible to remove insulation because of the type of insulation retainer system employed. This, of course, makes it very difficult, particularly in high radiation fields, to do any effective inspection work. A few minutes with a nondestructive test engineer, considering the necessity for inspecting the plant, can very well provide removable insulation at locations where the inspections are likely to be required.

We have found all kinds of structural steel, much of which is not shown on drawings, blocking areas such as the steam generator head, the piping, etc., and the nozzle to pipe juncture just off of the pressure vessel. Many of these structures were not designed with inspection in mind. As a matter of fact, many of these structures end up being placed after the components are in position and,

Figure 3. View Showing the Ultrasonic Module and Its Method of Centering in the Nozzle Bore

therefore, it is quite difficult to assure that a plant that is designed to be inspectable will be constructed to be inspectable.

ENVIRONMENTAL LIMITATIONS

The major environmental limitation to inspectability of plants is the radiation level. This, of course, varies with reactor design, type and power level as well as with the age of the structure. It is generally hard gamma radiation which is the most severe problem. This gamma radiation can be either shine from the core or in case the core is removed from induced radiation in the vessel shell. In other components such as piping, valves, pumps, heat exchangers, etc., the radiation can come from crud deposited on the surface which has been made radioactive by the radiation in the core and subsequently deposited on the surfaces remote from the pressure vessel. The major source of shielding for inspection operations is generally water which attenuates the gamma radiation. However, in many cases, radiation level can be reduced significantly by design of portable shields which can greatly increase the amount of time in which an inspector can be used without picking up a dose of radiation in excess of health physics levels. Of course, it is necessary to apply the proper health physics controls to inspection personnel. However, as long as the dose limits which are permissive by regulation are not exceeded, it seems that inspection can be properly conducted.

Crud deposits also may give difficulty in regard to inspection of reactor pressure surfaces. Because of the attenuation of ultrasonic energy resulting from this crud deposit, it may be necessary to resort to some cleaning operation which will provide a method of removal of such crud.

In summary, the inspection of vessels which had been in service can be greatly improved by consideration of the engineering requirements of inspection at the design stage of the equipment.

Detection of Leaks in Inaccessible Areas of the Boiling-Water-Reactor Nuclear Power Plant

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ABSTRACT

The detection of leaks from the nuclear steam supply system inside the drywell of a pressure suppression containment for a large boiling water reactor (BWR) is discussed. Instrumentation currently in use by the General Electric Co. is shown to give leak detectability to below 5 gpm of condensate using instrumentation such as flowmeters, temperature sensors, and pressure sensors.

1.0 INTRODUCTION TO THE BOILING WATER REACTOR

This paper will deal primarily with the detection of coolant leaks from the steam supply system inside the primary containment (pressure suppression system) of a large boiling water reactor (BWR). The detection of these leaks is somewhat complicated because during reactor operation the inside of the primary containment is inaccessible. The steam supply system piping is designed as a high integrity system. Any leaks which might occur will remain small for a long period of time, and it is important to be able to detect these leaks before they become a safety problem. The exact threshold of when a leak becomes a safety consideration has not been well established and no attempt will be made to establish that threshold in this paper. The emphasis in this paper is on detection methods that are currently employed.

Before discussing the various leak detection schemes, the primary containment design needs to be introduced. The function of the primary containment is to limit offsite radiation doses to negligible levels in the unlikely event of an accidental release of fission products. To accomplish this goal, the General Electric Co. recommends a pressure suppression type of containment for BWR plants. The BWR vessel, recirculation loops, and a large amount of related piping is located inside a leak-tight, steel vessel (referred to as the drywell). In Figure 1, an isometric view of a typical reactor building is shown. In the center of the picture, a cut-away portion of the drywell shows the reactor vessel and some of the piping systems which are located inside the drywell. The reinforced concrete shielding that surrounds the drywell is also shown. The

Figure 1. Boiling Water Reactor Building

large toroid-shaped vessel near the bottom is the pressure suppression chamber. In the event of a loss-of-coolant accident, the steam escaping from the primary steam supply system into the drywell is vented into the suppression chamber where the steam is condensed in a large pool of water. By condensing the steam in the suppression chamber, the volume of the primary containment can be substantially reduced for a specified maximum allowable pressure.

In Figure 2 the Dresden Unit No. 3 drywell vessel is shown prior to the erection of the reactor building. The size of the drywell vessel and the suppression chamber can be appreciated by noting the workmen in the lower foreground.

Finally, in Figure 3 an orthogonal sketch of the drywell is shown. To clarify the sketch much of the piping which is located inside the drywell has been omitted. Actually, the piping in the drywell is quite extensive; the minimum size of the drywell is controlled by the volume required to contain the necessary piping and the reactor vessel.

During reactor operation, personnel access to the drywell is prevented by nuclear radiation. Hence, leaks from the primary steam supply system which could

Figure 2. Dresden Nuclear Power Station Unit No. 3, Drywell Vessel

otherwise be easily detected visually must be detected and identified remotely. The detection of primary steam supply leaks inside the drywell is the main topic to be discussed in this paper.

2.0 INHERENT LEAK DETECTION CAPABILITY IN A PRESSURE SUPPRESSION CONTAINMENT

Because the drywell of a GE-BWR is a closed compact compartment containing the reactor pressure vessel and the greater portion of the primary steam supply system, a high degree of sensitivity to leaks is inherent in the design. Leaks

Figure 3. Cross Section of the Primary Containment of a GE Boiling Water Reactor

on the order of 3 gpm of condensate are detectable with little instrumentation beyond that normally provided for other purposes. Leaks on the order of 5 gpm cause so many detectable signals that detection of such a leak is ensured.

The high sensitivity to leaks comes from five sources:

1. Equipment Drain Sump
2. Floor Drain Sump
3. Drywell Ventilation System Cooling Water
4. Drywell Pressure
5. Drywell Temperature

2.1 EQUIPMENT DRAIN

First we will discuss the equipment drain sump which is located in the drywell. Most, if not all, of the normal background leakage is piped to the equipment drain sump. The collected water is pumped out of this sump and metered. All valves and pumps have double seals and normal leakage through the primary seal is drained off to a condensing pot and then to the equipment drain sump. Condensate from the cooling coils of the drywell ventilation system is also piped to the equipment drain sump. Normal flow to the equipment drain sump has been calculated to be on the order of 4 gpm so that an increase of 2 gpm would be readily detectable in a matter of hours. Since essentially all of a steam leak will be routed to the equipment drain sump as condensate from the ventilation system cooling coils, a 2 gpm steam leak produces a detectable signal. In the event of a leak from a near saturated liquid source, the water side of the primary steam supply loop, only about 40 percent of the leakage will flash to steam. Hence, a 2 gpm flow to the equipment drain sump requires a total liquid leakage of 5 gpm. Therefore, the equipment drain sump can detect 2 gpm steam leaks or 5 gpm liquid leaks as indicated at the top of Figure 4. In Figure 4 a marginal signal is indicated down to 1 gpm. A marginal signal is defined here to mean a signal which the operator should be able to detect, but the magnitude of the signal would fall within the range of possible background variations. In this particular case a 1 gpm increase in sump flow is a 25 percent increase which should be noted by the operator. But if all other systems are operating normally, the operator would probably attribute the increase to increased seal leakage. However, if there were also other indications of a leak the operator would be more inclined to attribute the increase to a leak into the drywell. The above philosophy will be applied in establishing marginal signals throughout this report.

2.2 FLOOR DRAIN

The floor drain sump receives all leakage not piped to the equipment drain sump. The collected water is also pumped out from this sump and metered. Flow to the floor drain sump should normally be quite low. Estimates are ranging from approximately 0 to 0.3 gpm. Because of the low background leakage, the floor drain sump is quite sensitive to small leaks. Assuming a 0.2 gpm increase in sump flow to be detectable gives a detectable total liquid leakage of only 0.33 gpm (recall 40 percent of the liquid leakage flashes to steam). In the event of a steam leak, essentially all of the leakage goes to the equipment drain sump as discussed above. Therefore, the floor drain sump cannot detect steam leaks.

Figure 4. Leak Detection Capability Inside the Primary Containment

2.3 VENTILATION SYSTEM

The drywell ventilation system is a water-cooled, forced-air system. The water in the cooling coils has a calculated temperature rise of 10°F under normal heat load and the air passing over the cooling coils is cooled approximately 30°F. On the order of 200 lb/h of steam is normally condensed on the coils, and varies considerably from plant to plant. By continuously monitoring the cooling water temperature rise (ΔT) across the cooling coils, an increase of 5°F should be easily detectable (15°F ΔT compared to 10°F normal). The increased latent heat load required to give a 5°F increase in ΔT is equivalent to about 3 gpm steam leak or a 7.5 gpm liquid leak.

2.4 TEMPERATURE AND PRESSURE

The drywell temperature (and pressure to some extent) is controlled by the drywell ventilation system. As the latent heat load on the cooling coils is increased, the average drywell temperature increases. The combination of increased temperature and increased absolute humidity causes the drywell pressure to increase. A 5°F increase in drywell temperature is considered detectable, which requires about a 3 gpm steam leak or about a 6 gpm liquid leak. A 0.5 psi increase in drywell pressure is considered detectable with the present drywell pressure monitoring system, and requires essentially the same leakage as a 5°F drywell temperature rise. The sensitivity of drywell temperature and pressure is also indicated in Figure 4.

2.5 LEAKAGE IDENTIFICATION

To identify the leakage as being from the primary system rather than from some auxiliary system, two signals are needed simultaneously. As shown in Figure 4 the monitoring scheme discussed above produces two strong signals for liquid leaks as low as 5 gpm and for steam leaks as low as 3 gpm. With two strong signals such leaks can be positively detected and identified as being from the primary system. Liquid leaks as low as 2.5 gpm and steam leaks as low as 2 gpm produce one strong signal and several marginal signals. Detection of these leaks is ensured by the one strong signal, and identification is highly probable because of the simultaneous marginal signals.

2.6 IMPROVED SENSITIVITY

The instrumentation previously discussed gives adequate sensitivity to primary system leaks. A leak of 5 gpm represents less than 10 percent of the control rod drive hydraulic system flow into the vessel, and under no circumstances would such a leak affect the coolant inventory in the vessel.

However, the sensitivity of the existing methods could be improved if the application warrants without the use of the more complex techniques to be discussed later. As stated earlier, condensate from the drywell ventilation system (Figure 5) is piped to the equipment drain sump where it is mixed with leakage water from various component seals. A flowmeter in the condensate line to the equipment drain sump (Figure 5) could detect much smaller changes in condensate flow than is possible by monitoring total sump flow. The flow rate in the condensate line would normally be approximately 0.4 gpm so that a 0.3 gpm increase in flow would be readily detectable. As stated earlier, a 0.3 gpm steam leak or a 0.75 gpm liquid leak would cause a 0.3 gpm increase in condensate flow.

A second signal which could be used is the dew point temperature at the air inlet duct of the drywell ventilation system cooling coil. Under normal operating conditions the dew point would not be expected to vary significantly. In the event of a leak from the primary system into the drywell the absolute humidity would increase and the dew point temperature would decrease. Industrial

Figure 5. Primary Containment Ventilation System

equipment is currently available which may be satisfactory for this application. Two such sensors are the vapor-equilibrium meter and the mechanized dew point meter.

2.7 VAPOR-EQUILIBRIUM METER

This sensor monitors the dew point by basically measuring the temperature at which lithium chloride (hygroscopic salt) is in vapor pressure equilibrium with the ambient air. To obtain this equilibrium temperature (dew point) an electric current is passed through the lithium chloride to heat it. When the dew point temperature is reached the moisture evaporates from the salt and the electrical resistance increases causing a drop in current.

This instrument has been used in industrial applications with sensitivity down to 1° to 2°F change in dew point. How the instrument would perform in a drywell type environment needs further investigation.

2.8 MECHANIZED DEW POINT METER

The mechanized dew point sensor employs a fog chamber transparent to light, which must be mechanically refrigerated. A light source shines through this fog chamber, and a photocell views the condensate (which forms at the dew point temperature) and controls the refrigeration. The fog chamber is thus alternately cooled and heated so that the condensate is made to appear and disappear in a cycle of 3 minutes. A temperature recorder of the thermocouple type must be used to record continuously the dew point temperature.

The complexity of this system is readily apparent, and its adaptation may not warrant the slight increase in leak detection capability.

It must be emphasized that these two instruments, the flowmeter in the condensate line from the cooling coils, and the dew point temperature monitor, are not currently employed in the drywell, because the added sensitivity which they provide is considered to be superfluous. They were discussed merely to point out the level of sensitivity which can be obtained without the use of the more complex techniques.

The preceding paragraphs encompassed only the detection of leaks inside the drywell. In addition, instrumentation is provided to detect leaks in the main steam supply line tunnel connecting the drywell and the turbine building. This instrumentation consists of several temperature sensors located at various points in the tunnel. The leak detection sensitivity of this monitoring system is difficult to predict analytically, but it is certainly in the range of less than 5 gpm.

In addition to the previously discussed methods of leak detection, the General Electric Co. has evaluated the practical applicability to nuclear power plants of leak detection methods currently employed in other industries. Because of the complex internal piping and environment of the drywell, these techniques do not appear to be feasible at this time.

3.0 CONCLUSIONS

The leak detection capability inside the drywell of the BWR pressure suppression containment is designed in such a manner that a leak of less than 5 gpm can be detected. Smaller leaks are not considered as detrimental to the safe operation of a large boiling water reactor. However, if future applications

warrant, sensitivity down to about 1 gpm is possible with additional instrumentation to monitor cooling coil condensate flow and dew point temperature.

Other detection methods currently employed in other industries do not appear to lend themselves to applications inside the primary containment of a nuclear power plant.

Panel Discussion:

Methodology for Building into Nuclear Power Plants Provisions for On-Line Testing and Inspection

Moderator: N. K. Sowards (PPC)

Panelists: W. G. Albert (Battelle—Northwest), W. C. Geckler (Holmes & Narver), G. H. Hake (AECL), D. R. Miller (GE), J. B. Nims (APDA), L. C. Oakes (ORNL), G. Stiehl (GA), and R. D. Wylie (SWRI).

SOWARDS: Out of consideration to the gentlemen on the panel who did not have an opportunity to give a paper earlier in this session, I would like for each of these individuals to say a few words relative to the general subject matter. After this we will open the discussion for questions or comments from other members of the panel or from the floor.

Mr. Geckler will now tell us how he relates reliability analysis and engineered safeguard systems to incipient failure detection.

GECKLER: I have listened to the papers that have been presented with two questions in mind, insofar as these papers apply to the area of work in which I have been involved. First, what does incipient failure diagnosis or detection mean to engineered safety systems, and, second, what relationships might exist between incipient failure diagnosis and reliability analysis?

It seems that the incipient failure diagnosis techniques are being developed in the area of anomaly detection in normally operating systems. In contrast, engineered safety systems by and large are constituted of equipment that is normally shut down. We are thus faced with the question of how to apply incipient failure detection to engineered safety systems of the normally nonoperating type.

Just thinking about what incipient failure detection is, if systems are developed and designed such that they give warning signals or actually perform trip functions in the reactor, then I see a whole new variety of protective systems being developed for reactors, and I wonder what this means in terms of the "noise level"—if you want to call it that—to which the reactor operator is already subjected.

As I see it, if we are going to use incipient failure detection techniques for engineered safety systems, the techniques will have to reflect failures in normally inoperative equipment, and that is going to be a real tough nut to crack. And, second, the techniques will have to reflect failures that may exist only under accident conditions, and this is an even tougher nut to crack. Even with the ability to test engineered safety systems in reactors that we now have, we are unable to test them under the true accident conditions.

Now, as to the relationship between incipient failure detection and reliability analysis, I think that there is a definite relationship that can be used to a very good effect. Mr. Balderston mentioned earlier that, before one does any incipient failure diagnosis, one should ask the following questions: What are the problems in making a system reliable? What are the failure modes? What are the causes of these failure modes?

One final comment about incipient failure detection might be that, in the broad sense of the subject, what is already being done on engineered safety systems, such as containment, emergency power, and emergency core cooling, is in effect a method of incipient failure detection. I offer that as a possible philosophic suggestion. If we look at this subject from the standpoint of the systems, these engineered safety system failures are not really important unless there is actually a demand present at the time the failure occurs.

In reliability analysis, and in the analysis of accumulated failure data to date, I wonder how incipient failure detection is going to handle the problem of failures that are induced by human error? I think that this is a problem that must be considered before extensive development of these techniques is begun because it might turn out that the techniques are not helping us very greatly if the human error problem is the thing that overrides the reliability of engineered safety systems.

SOWARDS: Mr. Stiehl, tell us whether these problems are present in the gas-cooled reactor business. This meeting so far has been highly oriented toward water reactors.

STIEHL: I want to talk not only about the gas-cooled reactors but also about ship power reactors. The Electric Boat Division of General Dynamics and also the Electrodynamics Division, which make submarine motors, have been doing research on how to reduce submarine machinery noise for many decades. In the October issue of *Mechanical Engineering*, there is an article that describes noise analysis as a maintenance tool on board submarines. It is a very simple, down-to-earth system. The navy machinist—the white-hat sailor—obtains a sound spectrum of a particular machine and compares it with a prepared overlay that has two limit lines. If the spectrum exceeds the first line, he starts making more frequent measurements. If the spectrum exceeds the second line, he notes the particular spectral bands that are out of tolerance, and he consults his handbook, which will give instructions to rebalance the motor, replace the bearings, or change the oil. This system depicts to me what our ultimate goal is for incipient failure detection.

I now want to turn to the Fort St. Vrain Nuclear Generating Station. It consists of an HTGR (High Temperature Gas Cooled Reactor) contained in a PCRV (Prestressed Concrete Reactor Vessel). The primary-coolant fluid is helium. The helium is circulated in the primary system by four steam-turbine-driven compressors. The helium is heated to 1400°F in the core. It then passes through heat-exchanger modules that generate 1000°F steam.

We have incorporated into the design two unique provisions for on-line testing and inspection. First, a minimum number of components are contained in the PCRV; hence, the vast majority of equipment is accessible for test and inspection at all times. For example, the PCRV tendons, which are the strength members of the containment, are accessible for inspection and retensioning, if necessary, at all times. The second is the primary-coolant moisture monitoring system. This system provides continuous measurement of the helium moisture content and provides the operator with prompt information on potential steam or water leaks.

SOWARDS: Mr. Albert will now say a few words relative to the problems for provision of testability on the Fast Flux Test Facility (FFTF).

ALBERT: The FFTF is in the conceptual design stage. One of the main concerns is to build into this reactor provisions for on-line system and component testing and inspection.

Today when a reactor design is started, one of the chief concerns is to get started on the design of long-lead-time items, and we all realize that one of these items is the pressure vessel. In order to obtain maximum flexibility in the design, we are trying to lay out the vessel so that we can incorporate innovations that the future might bring along.

For instance, if acoustic emission techniques can be developed some day to the point that they are of value in this reactor, we hope to have the system designed so that we can use them. To accomplish this objective of flexibility, we are making liberal use of inspection ports and are trying to keep the vessel interior as uncluttered as possible.

SOWARDS: Mr. Oakes will now comment on on-line testing and on the Molten-Salt Breeder Reactor (MSBR) program.

OAKES: I want to make several remarks about on-line testing in general before proceeding to the MSBR subject.

From our experience at ORNL, we have found that it is very difficult to design a "perfect" test system, and I think that the reasons why this cannot be done have been discussed in the Monday morning session. Also, after one has done the best design job one can do, one often finds that the cost incurred is very high indeed.

The question was asked here previously about what one might expect to pay for the provisions that allow testing of safety devices. I would estimate that the testing, engineering, and capital costs of the safety testing equipment for the High Flux Isotope Reactor (HFIR) were in excess of \$100,000. Approximately \$60,000 of this was for the hot water injection system alone. However, we feel sure that these costs can be reduced.

I would like to refer to Mr. Sowards' earlier comment that one cannot subject his reactor to a transient that is potentially damaging. I agree heartily with him and add that for precisely this reason we conducted tests several years ago at the SPERT facility in Idaho to establish the functional reliability of our vacuum-tube instrumentation in use at that time. We think that this particular series of tests gave us more assurance of the functional reliability of that instrumentation than any tests we have made before or since. It is unfortunate that we did not conduct the same tests with our new modular solid-state instrumentation.

At ORNL we are presently interested in molten-salt breeder reactors, and, although the design has not progressed into a very detailed stage, we have some concepts that have been fairly well established. The core will probably consist of two fluid regions: a fuel region and a fertile salt region. These regions will be separated by one or more graphite barriers, and the whole assembly will be contained in a Hastelloy vessel. One wonders if some of these incipient failure detection techniques might not be applicable to the problem of detecting minor cracks that might develop in the graphite. Knowing that the graphite is intact is of considerable safety and operational importance to us. Detection of small cracks would be very helpful. I have not heard any comment here about whether these acoustical techniques, for instance, might be applicable to graphite or Hastelloy; I assume that they would be. If some of these methods are indeed applicable, they could save us costly, undesirable vessel penetrations. The comment has been made several times here: "Why do we try to detect from outside the vessel?" The answer is that reactor vessel penetration is expensive and is looked upon with consternation by those who must see that fission products are contained.

In a different vein, I would also suggest that there is something of a "credibility gap" between what has been defined as existing technology and the actual engineering application of this technology. Considerable effort might be required to bridge the gap between laboratory application of these concepts and practical use of them in a system operating at 1200°F.

To conclude, I wonder how an operator might respond if, after running a system without incident for 4 to 5 years, he suddenly receives an alarm on an incipient failure instrument. Would he shut down his multimillion-dollar plant?

SOWARDS: I would now like to turn the microphone over to Mr. Hake for his opinions and comments.

HAKE: I have been quite confused listening to the papers given at this conference. People have talked about on-line testing as if it were something we did to prove the adequacy of the system design. I would hope that, before we place our power reactors into service, we would have proved the reactor design, and the testing we are talking about is thus the testing necessary to ensure that the performance of the system is still as specified in the design. So we should be looking for testing systems that check a sufficient number of functions to give us this satisfaction.

In Canada we are rather fortunate in that we have had availability figures given to us by our regulatory people. They have asked that our protective and control systems and our containment functions shall each have an availability (not a reliability) of 99.7%. If we go back to the first session of this conference, which involved the question of repair, we can relate this availability to the ratio of repair time and mean time to failure. For 99.7% availability, we would find that with a single system, using once-per-day test procedures that were 100% effective in detecting critical faults, we could have an acceptable fault rate of one fault every 2 years. But, if we had a triplicated system with the same testing frequency, then the individual availability of the systems could be less than 90%, and we could, in fact, tolerate about one fault every week.

Now, these are not unreasonable numbers, provided that we can ensure the redundancy or channel independence tacitly assumed. The bogeyman in Canada is not the fast fracture of pressure vessels; it is the man with the big wrench who drops it across these otherwise independent systems and makes them all common.

To make one suggestion, which may or may not be sensible: Since we will have digital computation available at the reactor site, I suggest that we can apply autocorrelation techniques to individual channels and cross-correlation techniques between the three channels to show that we do, in fact, have the required independence between the three channels. This would give us assurance that we are satisfying our availability objective.

SOWARDS: I would like to ask any of the panelists if they have questions for other members of the panel relative to the remarks just made or to one of the prior papers.

I will exercise the chairman's prerogative by directing the first question to Mr. Hake. You mentioned that we can actually determine a mean time between failure (MTBF) for these systems. For systems that we do not operate, like the safety systems in the normal plant mode, I think that we have a real problem. How can we obtain an MTBF for systems that we never run, on the basis of plant experience? Furthermore, if we could obtain an MTBF under a nonaccident environment, how do we have any confidence that this is meaningful in an accident situation? Maybe the environmental factor that we must stick onto these numbers in reliability analysis is zero for a key component.

HAKE: Well, I cannot of course answer the question of whether the system will work in an accident environment because I do not know how well you have done your design. I would suppose that you have simulated the accident circumstances well enough to be satisfied that the system would work when required. We then must choose test intervals that will give us a sufficient satisfaction that the system has not gone into some unsafe condition. I do not think that the MTBF should be looked at as sort of an absolute number. I think that you can use the ratio of the MTBF to the time of repair, with the time of repair being the time between tests. You can get a number that is practical for this situation by using a number for MTBF which you feel certain is going to be exceeded.

BALDERSTON (BOEING): I would like to propose an approach to the test of your safety items for incipient failure. We have run up against similar situations in testing the triggering of atomic warheads in nuclear missiles which we obviously cannot set off to check for correct operation.

What we must do first is to determine all possible failure modes of the system. Then we must look back and find the basic causes of those failures. In our considerable experience, incredible as it may seem, we found that in electronics failures and in mechanical, electrical, and structural failures, the basic causes were always limited to two: chemical contamination and structural defects. If one builds an incipient failure detection system or safety items with the idea of looking for these two basic causes, then one will get a good check of the system's operability.

It turns out, also, that one can often make these tests even better when the equipment is not operating than when it is. Sometimes this is difficult, and one has to go to exotic techniques to make such tests. I will cover the various techniques in my paper tomorrow, and so I will not go into those now.

Now there is one additional cause of failure I did not mention: human error. Human error can cause two different situations: The human can insert a failure mechanism that is incipient in nature, or he can cause immediate system failure. We can detect only the first of these with our incipient failure detection system.

DEBOISBLANC (IDAHO NUCLEAR): Mr. Hake, I would like to ask you to expand upon your suggestion for the use of autocorrelation and cross-correlation techniques to test for channel independence. How does one correlate the actions between a passive device, which is inactive until called upon to do something, such as a detector or an engineered safeguard, and an active device, such as the normal process variables of the reactor? This, I think, is the central problem of the engineered safeguard and its ultimate reliability.

HAKE: With an active control system, for example, I suppose that one could check a single channel for periodicity and make sure that there was none present. One could then insert a step function or some other periodic signal and see if it showed up in any of the other channels. If it did not, I would know that these channels are independent.

I do not know precisely how we would extend this idea to mechanical systems. It may be that with a dormant mechanical system one might only have to look at the very beginning of a motion, for example. Remember, I was only making a suggestion. Certainly the equipment and the necessary mathematics exist. It is thus a question of applying a concept to the real system.

GECKLER: I would like to qualify a previous comment that I made. Maybe with the advent of these ideas concerning incipient failure detection and an increased need to know if the whole engineered safety system is operable, we should seriously consider the concept of a continuously functioning safety system, as opposed to one that is normally dormant. In the past I think that one of the reasons we have shied away from using engineered safety systems in a continuous sense is that, if one operates them all the time and they finally fail, then one is confronted with the decision of whether the plant must be shut down because of a failure in an engineered safety system which has nothing to do with operation.

MILLER (GE): I would like to point out that, if one has a continuously running safety system, the reliability of those parts of the system which are exposed to the environments caused by the accident is not improved by continuously running other parts of the safety system. In the event of an accident, it is the sensors that are going to be exposed to transients and hostile environment, and having the core spray pumps running at the time of the accident is not going to improve the reliability of the sensors at all. Testing pumps periodically to make sure that they are functioning is a far better way to ensure that the pumps are available than having them running all the time.

SOWARDS: Maybe we should go one step farther and define the system so that it is passive with respect to the sensors and does not require an activated dynamic sensor system, if it is indeed a very crucial system.

HAKE: I mentioned earlier in the conference that there is available a description of the SCORE program, carried out by the Martin Company, on periodic testing and repair of stored missiles. This is in connection with Bullpup and Polaris missiles, which are dormant or in storage for very long periods of time. A periodic test technique is used to maintain the weapon system at a state of readiness, and some of these techniques might be applicable to our reactor situations.

SCHULTZ (MILLETRON): Can someone tell me what is current practice with respect to checking these large engineered safeguards? For example, if I have a core spray system inside a containment vessel, how often am I required to check this?

HAKE: I gave you the answer for Canada. Often enough to satisfy a 99.7% availability.

GECKLER: I cannot give you a number as to how often these safety systems are checked. In our recent evaluation work, we often wondered why, when, and how the systems were checked. It has been our experience that in those cases where tests were performed they were not done in such a manner as to give a great deal of confidence in the system.

DENTON (AEC): In most of the plants that are now operating, no credit has been given for engineered safeguards, and so a great deal of attention has not been given to their

testing procedures. But today the attempt is to treat the testing of engineered safeguards on the same basis as any other system. If you examine the technical specifications for the two plants that came on line most recently, Southern California and Connecticut Yankee, you will find that they state the frequency with which engineered safeguards are tested and give reasons why the stated test frequency was chosen. I do not remember the testing frequencies offhand.

BALLARD (SANDIA): I would like to point out that in the nuclear weapons operation and maintenance business we have problems exactly analogous to reactor engineered safeguards testing. In the early days of the first nuclear weapons, we could afford to put a Ph.D. with each weapon to give it tender, loving care, to take it apart, and to put it back together. However, when these weapons are built by the thousands and put in silos and in Polaris submarines, with maintenance being performed by men with limited technical training, the whole reliability picture has to take another viewpoint. I see some similarity between this weapon situation and the forthcoming proliferation of power plants, and it is time, I think, that some of the rationale that has been applied to things like dangerous weapons be applied to nuclear power plants.

As a new approach to the reliability problem, we started devoting most of our effort toward the type of failure mode analysis described by Mr. Balderston. That is, we started analyzing the various detailed modes by which these weapons could fail or detonate prematurely. In light of this analysis, we decided to perform very exacting tests on a sample drawn at random from the entire weapons stockpile. These tests begin with a component-by-component dismantling, followed by a number of sophisticated analyses.

I feel that it is more important to do a sophisticated job of analyzing the modes of failure than it is to do repetitive testing. By applying enough redundancy to the system, as mentioned by Mr. Hake, we can have a component fail every now and then without jeopardizing the reliability of the system.

OAKES: I would like to comment on Mr. Ballard's remarks. One assumes that all of the atomic weapons in our arsenal are very similar in nature, so that one could obtain a worthwhile statistical sample by extracting one or a few of these bombs at random.

Unfortunately, we have very few similar nuclear reactors, and they are sufficiently different, I think, that doing the job described by Mr. Ballard would require disassembly of each and every reactor piece by piece.

BALLARD: No, because I assume that the safety system portions of those reactors would be very similar; at least I gather that there are enough similar sensors to allow one to do some of this. And I also point out that nuclear weapons are not all alike. We make big ones, little ones, ones that will stand shock, ones that will work under water, etc.

MILLER (GE): I believe that, if the scope of this discussion be limited to nuclear power plants rather than nuclear reactors in general, there is found far less variety in control and safety systems. The BWR's to be built by GE in the future will be basically the same as those being built now, with minor modifications on a slowly changing basis, and I am sure that this is also true of all PWR manufacturers. I do not think that we are going to a future situation where every nuclear power plant is different.

SOWARDS: A final question to Mr. Wylie. Someone has already mentioned that one of the problems facing us in applying the stress-wave acoustic emission techniques to reactors is that we lack a criterion for how good a flaw detection system we really need. That is, what constitutes an unsafe vessel condition, or what flaw size is significant or insignificant? We need to know answers to these questions or how they can be obtained.

WYLIE: One cannot give a unique answer to this question because each component probably has a different permissible flaw size, depending on the stress it must withstand, the number of cycles, the operating history, etc. To a certain extent, the answer is also dependent on what one can tolerate in the way of failure. This is a definition of what is "incipient," you see.

Much work is going on with regard to this particular problem, and I mention the HSST program managed by ORNL which is directed toward a better understanding of permissible flaw sizes in heavy-section pressure-vessel steels. Similar work is being done in analysis of permissible defect sizes in piping.

Hopefully within the next few years one will have enough information to select a criterion for decision-making on what flaw size we can safely tolerate.

**Incipient Failure Detection and Malfunction
Diagnosis Methods Applied in Fields
Not Related to Nuclear Reactors**

W. W. Mutch, Chairman

Incipient Failure Detection: A System Concept

H. L. BALDERSTON
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SUMMARY

The detection of incipient failure of a part, component, subsystem or system is related to the detection of latent defects which already exist in a device, develop during operation of the device, or appear as a result of operational environment, or human error. Defects may also appear as a result of unexpected stress or other unexpected operational hazards. Incipient failure detection is directed to the most basic causes of failure, that is, structural defects or chemical contaminants. Further, these two basic causes of failure apply to structural assemblies, mechanical assemblies, electrical and electronic assemblies. Specifically, the detection techniques may include the use of vibration, acoustical energy, electrical noise, infra red, ultra violet, x-ray and gamma radiations. This paper will indicate how existing knowledge and new research can be combined to develop an incipient failure detection subsystem with broad applicability to aerospace vehicles and ground equipment installations.

On December 1, 1965, the BOMITE (Boeing Microelectronic Integrated Test Equipment) concept developed by the author had been completed. BOMITE was an integrated test equipment concept designed to be built into, and made a part of, operational equipment and to give that equipment a self-test-for-failure capability. At various times during the development of the concept it was observed that tests of impending failure would have been even more valuable. In fact, a file was made covering techniques which might be usable and work done by others in nondestructive testing, anticipatory failure detection, physics of failure and other areas wherein usable techniques had been or were being developed. In electronics the detection of incipient failure appeared to be limited to discrete parts and was limited to techniques usable in sorting "mavericks" or "deviates" from "lots" on large numbers of similar parts. Information was accumulated on sorting techniques applicable to resistors, capacitors, diodes, transistors, field effect transistors and selenium rectifiers. Electrical noise appeared to be the tool most widely employed in the above techniques. In structures, vibration, ultrasonic, acoustic emission, x-ray, and infra red radiations

appeared to be most often employed, in mechanical equipments sonics and ultrasonics were the most commonly used tools. It appeared that noise in its widest definition including vibration, acoustical, electrical, infra red, ultra violet, x-ray and gamma radiations would provide the techniques necessary. It was planned to utilize existing techniques wherein possible, modify existing techniques wherein necessary and develop new techniques to fill in the gaps to enable the development of an incipient failure detection subsystem.

GENERAL DISCUSSION

The existence of noise (vibration, acoustic, electric, infra red, x-ray, ultra violet and gamma) of one sub-classification or another commonly occurs in all devices and structures. Vibration is common in structural and mechanical devices. Acoustical noise and vibration is common to rotating, reciprocating and moving mechanical or structural equipments. Electrical noise is common in electrical and electronic equipments. The infra red, x-ray, ultra violet and gamma sub-classifications may appear naturally and are quite useful when injected artificially. The existence of "noise" to varying degrees and of various sub-classifications may be expected. It is possible, by careful design and selection of parts, to eliminate some noise sub-classifications and/or frequencies and to hold others to acceptable levels. When such equipment, so designed, is placed in operation and properly maintained noise should remain at acceptable levels throughout its trouble-free operating lifetime. The existence of abnormal quantitative amounts of noise at any time during operation of equipment indicates first a deviation from normal such as a lack of oil on the bearing surfaces of rotating machinery, the existence of a destructive electro-chemical process or it may indicate excessive wear, cracks, voids, discontinuities, an out of balance condition, or other internal damage.

When other noise sub-classifications, not normally present, begin to appear, such as vibration or excessive electrical noise, severe abnormal conditions may be indicated. In either event it has been shown that the existence of abnormal quantitative amounts of noise and/or the introduction of additional sub-classifications of noise, may indicate a risk of failure, that is, a defect, or a failure mechanism in progress. Thus, in incipient failure detection, we are interested in quantitative amounts of noise and the introduction of noise sub-classifications or frequencies not normally present. Let us examine the various noise types and the possible implications of their appearance in abnormal quantitative amounts.

ACOUSTICAL NOISE (VIBRATION, SONIC AND ULTRASONIC)

In rotating machinery the sounds produced are usually audible, however they may extend above the range of human hearing and are produced by rotation and/or vibration. Vibration (piezoelectric) type transducers are normally employed for monitoring such sounds. In rotating machinery any defects, wear or damage creating an out of balance condition to the rotating mechanism increases audible sound, commonly classified as due to vibration, which builds up in intensity as these incipient failure conditions progressively increase. With acoustic transducers properly installed it is possible to detect the existence of these defects well in advance of failure. The techniques of instrumentation, data recording and analysis are well known and extensively documented. Various equipments such as that built by Spectral Dynamics Corporation and by Curtis Wright are available. (Bibliography - 1 through 9.)

In structures under stress or subject to bending moments metals may reach stress limits wherein local plastic yielding occurs. At this time metals may emit a

"frying sound" which usually cannot be heard unaided by the human ear but with the proper transducers and sufficient amplification the "frying sounds" become audible. The sounds have been described as similar to that of bacon frying. The phrase "metals cry under stress" has been employed. The monitoring of this noise phenomenon called acoustic emission has been useful in defining areas of excessive stress and even in predicting failures in pressure vessels and other structures. (Bibliography - 10 and 11) Emissions, above the range of hearing in the range of 100 KC to one megacycle, have been observed. Studies at Boeing Scientific Research Laboratories have shown the existence of electron and photon emission but much additional research will be required to utilize these latter two emissions.

Ultrasonics has been shown to be an effective tool in the detection of voids, cracks and discontinuities, changes in crystal structure, and local plastic flow in structural assemblies. The techniques involve the application of ultrasonic energy, usually in pulse form, and the monitoring of return pulses against time and amplitude. When return pulses are monitored against time, the location of the crack or discontinuity is indicated or the thickness of the material or both. The height of the return pulse is an indicator of the extent of the crack and the impedance of the material through which the acoustic pulse has traveled. An alternative method involves the monitoring of the amplitude of the pulse to determine transmittance of the acoustic wave through the material and/or impedance of material to the propagation of acoustic energy. These last two techniques may be especially valuable in the determination of changes in crystal structure (fatigue) of the material or local yielding of the material. The resonance technique which is employed for determination of thickness and which detects thin spots in the walls of tanks or structural members, involves the use of special wide band transducers and tuning to resonant frequencies of the material thickness. That is, when resonance occurs, the wave length of the ultrasonic wave applied to the material is an exact multiple of the material thickness or is equal to the material thickness, thus the instrument can actually be calibrated in material thickness in lieu of wave length or frequency. In the multiple pulse echo technique more accuracy is achieved than with single echoes since multiple echoes from bottom surfaces of the material are employed. The thickness is then equal to the screen length covered by the multiple echoes divided by the number of echoes viewed on the screen. In the pulse echo techniques, Krautkramer Ultrasonics, Inc., Automation Industries, Inc., and the Magnaflux Corporation all have done considerable work and market equipment usable in this technique. In addition, Automation Industries has attachments which will measure transmittance and/or impedance directly. The other manufacturers mentioned may also have these attachments. Krautkramer Ultrasonics, Inc., also has an excellent, fully transistorized battery operated unit with built-in alarm system and attachable miniature strip recorder. These equipments are excellent for walk-around tests of defects in structures or structural components. Tests for voids, cracks and discontinuities in the metal, either longitudinally or transversely, may be made with ultrasonic equipment. In addition, welded joints can be inspected for cracks, voids, and discontinuities by the use of transverse ultrasonic waves. Krautkramer Ultrasonics, Inc., also produces high speed automatic test equipment which runs tests on tubes, bar stock, ingots, sheet stock, etc., continuously at very high speeds while the material is transported by or through the test stand. Extensive employment of this automatic equipment exists in Europe in the manufacture as well as in the assembly of these products in pipe lines, boilers, chemical processing plants, petroleum plants, large and small structures, automotive and aircraft plants, etc. It is certain that more extensive use of these techniques will be developed in the immediate future. (Bibliography - 12 through 22.) In addition, new uses for ultrasonic energy are being announced regularly in trade journals and industry literature. Last month, for example, the Chesapeake Instrument

Corporation of Shadyside, Maryland, announced the development and marketing of an ultrasonic analyzer which was capable of measuring and/or controlling the concentration of solutions. Typical chemicals which are being measured or controlled are solutions of glycerine, sodium hydroxide, alcohol, acetone, and butterfat in milk. This equipment may be utilized to check the extent of contaminants and viscosity in oil and other fluids. (Bibliography - 23 through 25.) The author is currently working on techniques utilizing ultrasonics in the detection of defects in the structural assemblies of electronics, including micro-electronics. This involves not only the packaging but also the inter and intra-connections, the functional electronic blocks down to and including micro-electronic integrated circuit chips. These techniques will be usable, hopefully, in all types of electronic subsystems.

The Fatigue Life Gage developed and patented by D. R. Harting of The Boeing Company has been shown to be effective in the detection of fatigue and in determining remaining life of the structure under test. The Fatigue Life Gage is a small, bondable resistance sensor similar in appearance to a foil strain gage. When installed on critical areas of a structural component which is subject to fatigue loading, the Fatigue Life Gage generates a permanent resistance change as a continuous (cumulative) function of fatigue experience. It can be interrogated periodically to read out its cumulative resistance value and thereby determine the remaining life span of the structural component when placed at the point of stress. It is manufactured under Boeing agreement to Micro-Measurements Inc. (Bibliography - Number 26.)

ELECTRICAL NOISE

Since electrical noise is inherent in any electrical device, it is necessary to accept certain levels of electrical noise. It is also expected that the quantitative amount of acceptable noise will vary from one type of device to another. The existence of certain types of noise in abnormal quantities has been shown to be indicative of a defect or characteristic leading to failure.

APPLICATION TO ELECTRONIC PARTS

The work which has been accomplished so far and published, relating abnormal noise to incipient failure is related mostly to resistors, capacitors, transistors, field effect transistors, diodes, selenium-rectifiers, and integrated circuits. Most of this work, and available equipments from Quan-Tech and Dale is directed toward the detection of "deviates" or "mavericks" among lots of the same part in sorting techniques. In such techniques, to eliminate the parts with high probabilities of incipient failure, a relatively large number of good devices must be discarded along with the bad. The basic structural defects which contribute to failure are chemical contamination and structural defects (voids, cracks, and discontinuities). The presence of a contaminating process develops noise through the ionization process which is detectable with noise evaluation equipment. Structural defects (voids, cracks, discontinuities) increase the basic noise level, or "excess" noise level since the opposition to a current flow through a void, around a void, a crack or discontinuity causes a voltage drop due primarily to an impedance increase at that point. (Bibliography - 27 through 34.)

INCIPIENT FAILURE DETECTION IN OPERATIONAL ELECTRONIC SUBSYSTEMS AND/OR COMPONENTS

The sorting techniques previously covered, along with quality control line inspection, functional test, and the many quality assurance tests which are normally conducted during the manufacture of electronic components and sub-

assemblies, should assure a high probability of producing reliable components and subsystems. But they are not perfect; accordingly, one can never be certain that no probability of incipient failure exists in an electronic component or subsystem. Therefore, it is desirable to have the capability of detecting an incipient failure once these components and subsystems have been placed in operational employment. In addition, incipient failure can result from operation of electronic subsystems and components in hostile environments and through human error. For all these reasons, then, incipient failure detection will prove to be a valuable tool leading to higher overall system reliabilities in operational airborne and ground operational systems.

While a considerable amount of work has been done in the development of incipient failure detection techniques for electronic parts, very little work has been completed in the detection of incipient failure in electronic subassemblies and subsystems.

THE DETECTION OF ELECTRO-CHEMICAL CONTAMINATING PROCESSES IN OPERATIONAL ELECTRONIC COMPONENTS AND SUBSYSTEMS

The author has just completed work and has published a paper covering the detection of four separate contaminating processes involving integrated circuit electronic assemblies. (Bibliography - Number 35.) These processes were found to be detectable during operation by monitoring the electrical noise content, in various parts of the circuit, at 10 cycles per second and at 1,000 cycles per second using a one cycle bandwidth in each case. The contamination process itself was observed to be a start-stop reaction. As failure became imminent the process became continuous and electrical noise increased from 10 to 100 times normal. At this time a prediction of imminent failure could be made and was made in each and every case wherein a failure occurred. The time of failure could not be predicted; only the fact that failure was imminent, imminent in this case meaning hours, or days, or weeks.

During these experiments we also observed that some structural defects were detectable through the use of electrical noise. It is doubtful that electrical noise will prove effective for detecting structural defects because of the difficulty of differentiating between high noise readings normal to some circuit paths from equivalent noise readings indicating structural defects in others. On the other hand acoustical noise, covered in a previous section, may prove to be an effective tool in the detection of structural defects. However, a great deal of research will have to be done to perfect usable, practical detection techniques. We are working in these areas and will publish reports if and when usable techniques are developed.

THE EMPLOYMENT OF ELECTRICAL NOISE AT MICROWAVE FREQUENCIES

It has been shown that the employment of electrical noise at microwave frequencies may be an effective tool in the detection of surface structural defects in metallic materials and defects in depth in plastics. (Bibliography - Number 36) These may be usable in some electronic assemblies. Research would have to be employed in this area to determine whether or not this category of electrical noise would be effective in incipient failure detection of electronic assemblies.

INFRA RED RADIATION

Infra red radiations are currently being employed in the laboratory to scan micro-electrical and electronic circuits for so-called "hot spots". These hot spots are indicative of areas of high resistivity; that is, structural defects. For example, in the case of a film resistor, if in scanning along the length of

the resistor one finds a hot spot it is indicative of a variation in the resistivity of that area and, as such, constitutes a structural defect which has a high probability of leading to failure. (The Philco Sierra Electronics Division 710-A Infra Red Scanning Equipment is an example.) Another technique recently developed by AGA AB, Poslags, Näsby, Sweden, and trademarked "AGA Thermovision" utilizes infra red cameras which provide a direct presentation of a thermo picture on a cathode-ray tube. Possible applications of this system include the detection of hot spots in high voltage power systems and other large electrical assemblies, indicating structural defects at such points. For other examples, see Bibliography - 37 through 42.

A program employing SiC Whiskers is being carried on by Richard F. Holtman and Lee W. Olson of The Boeing Company. Technical Report Number RADC-TR-67-273 dated July 1967 was the final report of this work. Considering the cost of these whiskers, their microminiature size and their sensitivity, a continued research should provide a number of applications in the detection of structural defects through monitoring of infra red radiation. These techniques are dynamic, that is, monitoring can be accomplished while the circuit is in operation.

Many different types of stress coatings (Stresscoat by Magnaflux, for example), liquid crystals developed by Wayne Woodmansee of Boeing, paint and lacquers are available which will indicate stress and surface structural defects, including voids and delamination, when activated by infra red, visible or ultra violet radiations. While these techniques are usable in the laboratory, in quality control inspection on production lines, their use in an incipient failure detection subsystem requires additional application research.

VISIBLE AND/OR COHERENT LIGHT ENERGY

Techniques employing visible and/or coherent light as from a laser beam are employed in techniques designed to evaluate surface defects in structures and structural materials. Laser beam holography, as developed at The Boeing Company, is particularly valuable in detecting surface defects, separation of surface layers from internal honeycomb structures, etc. Additional research will be required to adapt this technique to an incipient failure detection subsystem. It can be used for production testing, and quality control inspection in its present form. (Bibliography - Number 43.)

ULTRA VIOLET RADIATION

Ultra violet radiation can be employed with a number of fluorescent stress coatings such as Heatmap, a product of Shallander Dev. Company. An electronic assembly is coated with the fluorescent paint by brush, spray or dip. The circuit is operated and is viewed under an ultra violet lamp. Hot spots are indicated as dark areas. (Bibliography - 44.)

X-RAY AND GAMMA RADIATION

We anticipate the necessity of using x-ray and gamma radiations in those cases wherein ultrasonics proves impractical because of the existence of air or other gas in a test path. In walk-around detection techniques these radiations in hand-carried x-ray generators or gamma sources may be quite useful. However, the use of x-ray and gamma must be limited to areas wherein no damage will result to radiation sensitive components, or to chemical compounds, (solid booster materials), which will undergo chemical change due to radiation. (Bibliography Number 45.)

CONCLUSION

The application of the techniques we have briefly covered to an incipient failure detection subsystem will require instrumentation, data collection, and data analysis. At Boeing we believe such a program to be feasible. Accordingly, we are currently performing research and starting development work leading to the development of such a system. We intend to instrument one aircraft, doing the necessary research and development as we go.

We will start by instrumenting an engine, developing the necessary techniques of instrumentation, data collection and processing. At the same time other rotating equipments will be instrumented. We will use sonic techniques for the rotating mechanisms and ultrasonics for the cases, assemblies and brackets. We will use ultrasonics for the structure in walk-around techniques wherever possible. In addition, x-ray, gamma, acoustic emission and possibly ultra violet and coherent visible radiations may be employed as applicable. It is our plan to instrument a hydraulic system in the laboratory using sonics and ultrasonics and if successful then instrument the system aboard the aircraft. In the meantime various staffs will be doing the necessary research leading to the instrumentation of the electrical and electronic subsystems. The total job may take from three to five years to reach a point wherein we consider the concept usable. Once such a system is installed and is in use by the commercial airlines a free exchange of information, with modifications, additions and changes, the system should develop until a very satisfactory percentage of all potential failures are detected prior to actual occurrence. We know that if some incipient failures are detected early enough failure can be prevented from occurring. In any event our goal will be to detect incipient failures in time to take effective action to prevent, control or eliminate actual failure.

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Discussion on Paper by Balderston

MILLER (ORNL): Mr. Balderston, you have talked at some length on the virtues of these various nondestructive tests, including ultrasound. Have you any suggestions for the ultrasonic examination of very coarse-grained austenitic stainless-steel welds in heavy-section vessels?

BALDERSTON: Krautkramer Ultrasonics, Inc., or Automation Industries have full information on the techniques that you would utilize. But maybe I misunderstood your question. Are you assuming that there is some special problem with stainless steel compared to other types of welds?

MILLER: Yes, simply because of the coarse-grain structure.

BALDERSTON: I do not understand how grain size affects the method, but, even if it does produce scattering, all one loses is a little energy. One always has lots of energy. One should still get echoes back.

Use of Television for Space Vehicle Testing

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As early as ten years ago there was a need in the missile field for pictorial confirmation of events occurring during testing and flight of the various vehicles. The first use of television was on a flight vehicle in 1961 to view the separation of two stages of the vehicle. Sophistication of the vehicles at that time had not progressed sufficiently for our test people to feel that the use of television was warranted. With the coming of the Saturn, however, the use of television was again employed beginning in 1963 for five consecutive flights to observe various onboard functions that occurred. Engineers involved with the testing, inspection, and assembly of the static vehicles realized from the first of these flights that television was a tool which could be profitably used in their areas as well. This will be a presentation of the current uses of television for testing within the George C. Marshall Space Flight Center, National Aeronautics and Space Administration.

Figure 1 is the high G centrifuge used to check various flight components for high acceleration. In the center of this picture is the television camera. The entire camera is mounted on the center axis of the spin table and pointed so as to observe specimens at the end of the spin arm nearest the observer. It is equipped with remote controls for a zoom lens and focus, thus allowing the specimen attached to the end of the arm to be viewed remotely from outside the test cell. The camera controls for focusing and zoom as well as the video signal are brought out by way of slip rings. Use of the slip rings does not contribute any noise to the picture. The specimen shown in Figure 1 attached to the end of the spin arm is a capillary matrix, two kilowatt fuel cell.

An application combining specialized optics and a TV camera is currently being used to check certain areas of the outer tank walls of stages for possible fuel leaks. The observations must be made at a remote distance. For this purpose a Questar telescope was combined with a 2.54 cm vidicon camera and tested under the following conditions.

The camera was mounted directly to the telescope eyepiece with a 3.81 cm adapter barrel. This arrangement is shown in Figure 2. This

Figure 1 High G Centrifuge (ASTR)

Figure 2 Questar Telescope with TV Camera Attached (TEST)

device was located inside the building at point A in Figure 3 and focused on point B. At point B a section of a tank wall and outside insulating material was set up in the open field. The distance between the TV camera and the wall sample was 137.16 meters. A pencil mark was made on the insulating material of the wall sample. When viewed by the

Figure 3 Questar Test Location (TEST)

specially equipped camera at point A, the image on the TV screen was as shown in Figure 4 with the black pencil mark showing in the center of the photograph.

A hole was punched in the insulation and liquid nitrogen was forced through from the back side of the wall simulating a fuel leak in the tank walls. The nitrogen leak is shown in Figure 5, but the puncture in the wall is obscured by the fog created by the expanding gases around the hole itself. In order to see the general shape of the hole, the camera was removed and the pickup tube in the camera was replaced with a tube sensitive to infrared. The camera was replaced in the optical assembly and again focused on the leak in the tank wall. With the infrared tube in operation, the image appeared as shown in Figure 6. The hole in the wall, being the coldest point, produced the dark L-

Figure 4 Pencil Mark at 137 Meters (TEST)

Figure 5 Fuel Leak at 137 Meters with Standard Camera (TEST)

Figure 6 Fuel Leak at 137 Meters with IR Camera (TEST)

shaped pattern while the rapidly expanding gases, warming up as they come from the hole, produced a white area surrounding the L-shaped hole. This has worked quite well for viewing defined areas of probable leak.

In one laboratory there is a special acoustical test facility for checking reaction of sections of the vehicle to high sound levels. Since the sound levels employed are considerably in excess of that which can be tolerated by a human, the operation within the test room must be monitored by a TV camera. This camera must withstand the high acoustical level encountered within the test room. This level may be as high as 170 db. The camera shown in Figure 7 satisfies these requirements. It is equipped with a pan and tilt unit, zoom lens, remote focus and iris controls. This camera is located in the reverberation test room. The source of the noise for this test room is a very large exponential horn, Figure 8, which is 1.22 meters across and 2.44 meters high at the mouth. With this horn the room has a low frequency cutoff of about 30 hertz.

Figure 9 shows another view of the room with a specimen to be tested. The specimen is a section of a fuel tank wall mounted as a part of the wall of this room and is shown on the left side of Figure 9. On the other side of this tank wall is another room called an anechoic room or "dead" room. A high sound level is generated in the reverberation room and the sound levels are mapped by microphones suspended on three sets of lines from top to bottom of the photograph. The sets are located along the right side of the panel, down the middle, and just out of view at the left side of the picture. An instrumentation arrangement on the anechoic side of the tank wall maps out the sound area on

Figure 7 TV Camera used in High Sound Level Chamber (P&VE)

that side in the same manner. The difference between these two sound fields is the energy absorbed by the tank wall and is that which has been taken up by motion. As an additional aid, accelerometers are mounted on this panel to verify the existence of specific levels at certain points on the tank wall. A-12 in Figure 9 is an example. The device in the upper left of the picture is an impedance shaker, which may be attached to the test specimen and used to run a sweep frequency vibration test and record the mechanical impedance of the wall at a point. This test checks on the wall mounting and its response characteristics. For this test the camera is used to observe the shaker to insure that it does not become detached from the wall, since it is a delicate instrument and subject to occasional detachment. The door of the room is closed to confine the sound level to the chamber and keep out any external noises from entering the chamber. A TV camera is also mounted in the anechoic room on the opposite side of the test wall to monitor the test wall conditions during such a test.

The most extensive use of closed circuit TV in our center is within the Test Laboratory. Figure 10 shows the S-IC test stand. Shown in the stand is the S-IC-3 flight vehicle. In this stand the test firing of F-1 engines is carried out at a total thrust of 33.5×10^6 N. Such a

Figure 8 Horn to Produce High Sound Levels (P&VE)

Figure 9 Test Setup in High Sound Level Chamber (P&VE)

Figure 10 Test Stand for F-1 Engines (TEST)

complex operation as this employs a number of television cameras for monitoring the various operations connected with the engine firing. Figure 11 shows the lower section of a single F-1 engine. Five such engines are mounted within this test stand. Three types of camera systems are employed by the Test Laboratory. One type is a high environment housing containing a cylindrical camera using a 2.54 cm vidicon. This housing is shown in Figure 12. The camera is surrounded by foam rubber within the housing and is mounted on a frame suspended from the floor of one of the test stand levels. Three cameras mounted by this method observe the upper section of the engine which would otherwise be inaccessible for observation. These cameras show any loose components, the icing of fuel lines, and the flutter or motion of components during the engine firing.

Another type of housing is explosion proof. This is the cylindrical housing shown in Figure 13. At the left end of the camera can be seen the purging line which keeps the inside pressure positive using

Figure 11 Bell and Lower
Section of F-1 Engine (TEST)

Figure 12 Heavy Duty TV Camera Installation (TEST)

Figure 13 Purged and Medium Duty TV Cameras (TEST)

dry nitrogen. Figure 13 also shows a square housing which is considered the intermediate duty housing and is used in those areas where explosion proofing is not required and where the vibration and acoustic levels do not call for the use of the heavy duty camera shown in Figure 12. A typical view of the explosion proof camera installed and the general area of observation with regard to distance between camera and subject material is shown in Figure 14. This picture was made on the S-IVB Battleship test stand which is the liquid hydrogen test stand for the J-2 engine. The camera viewed such things as the upper part of the engines, the turbo pump mechanism, the suction lines, the various control packages, and the high pressure speheres.

Another TV application in the test area is on the swing arm test stand. The swing arms used for launch to carry the umbilical cables between the launch tower and the vehicle on the launch pad are tested here. The arm is put in place on this test fixture and all of the lines which the arm will carry, including fuel, operating, and control lines of any type, are connected to the arm. Then vibration of the vehicle due to engine thrust from liftoff and wind loading on the vehicle are simulated and the arm is activated to swing away from the side of the tower. A TV camera observes this operation and its output is tape recorded. This recording can be immediately played back for the test engineers in slow motion and stopped frame to insure that the separation of the arm was correct. These same arms, after passing the test, are shipped to the launch site and installed on the umbilical tower for the actual launch operation of the vehicle.

Fiberoptics have been used as coupling devices between the specimen and TV cameras as well as movie cameras. The fiberoptics that have

Figure 14 View of TV Camera and Upper Section J-2 Engine (TEST)

been used are composed of glass threads sixty microns square. Each thread in turn is composed of 36 ten micron individual threads fused together to form one bundle, but having 36 individual light paths. The thread is wound in layers to form the bundle which is cemented together to hold the thread position. The bundle is then cut across the cemented joint. When the ends of the cut are polished, there is provided a coherent light transmission path between the two ends of the bundle. These fibers give a resolution through the bundle of 55 lines per millimeter in both planes. The format employed is 9×12 mm or the size of a 2.54 cm television camera tube image. This fiber bundle is wrapped in teflon and then encased in a steel hydraulic cable. A lens may be attached to one end producing its image on the polished face of the fiber bundle. It has been found best to use a two lens system at the opposite end. One lens system acts as the collimator to convert the image to a parallel plane while the second lens system converts the plane image to a focused image on the face of the camera tube. Thus, the system may be operated with as much as ten cm between the two lens systems at the camera end of the fiberoptic bundle. Figure 15 shows the installation of a lens assembly with a fiberoptic bundle and incandescent light assembly on the bulkhead of a liquid oxygen fuel tank of a vehicle. The lens assembly in this case has a dry, warm nitrogen gas flow passing through the housing to prevent fogging of the lens and faceplate in the top of the fuel tank.

Another application of the fibers is shown in Figure 16 where the lens housing is pointing upward to view the separation of a second stage. The cable for the fiberoptics comes around the bottom of the tank assembly ring into the camera housing. When the stage separates, a picture is obtained like that shown in Figure 17. This shows the

Figure 15 Fiberoptic Bundle Mounted on Fuel Tank (ASTR)

Figure 16 Fiberoptic Lens Housing Location on top of Saturn Stage (ASTR)

Figure 17 Fiberoptic View of Separation of Saturn Second Stage (ASTR)

second stage containing six engines at a distance from the opening in the top of the first stage of approximately 30 meters. Three of the engines have already built up heat sufficient to show while the other three are still coming up to temperature. This picture was made from a single frame of sixteen millimeter format size.

A second method for observing fuel in the tank employs a TV camera located on the tank bulkhead in a special housing and purged similar to the lens assembly in the fiberoptic bundle shown in Figure 15. Two cameras were mounted in the top of the fuel tank with this sealed housing purging arrangement along with two incandescent light assemblies which illuminated the inside of the fuel tank. The cameras were connected to transmitters on board the vehicle to transmit pictures of fuel action under zero gravity conditions to earth-based stations. The second stage of the vehicle containing residual fuel was placed into orbit and various conditions of acceleration and the absence of it were observed and recorded at the earth-based stations. In Figure 18 the dark area of the picture shows the bottom of the fuel tank with the fuel broken up into various shaped areas and beginning to move toward point A on its way to the forward end of the tank. This is one frame of the television picture as received on monitors on the ground. These tests were conducted to prove the theories of action and reaction of fuel under orbital conditions.

When the micrometeoroid satellites, the Pegasus, were to be put into orbit, a requirement was placed on our laboratory to provide a camera to view the deployment of the wing assembly to determine if it unfolded properly and locked into position. Figure 19 shows the approx-

Figure 18 Liquid Oxygen in Fuel Tank under Zero Gravity (ASTR)

Figure 19 Saturn TV System with Pegasus Stored (ASTR)

imate location of the camera relative to the stored Pegasus system. The view of the TV camera was toward the nose of the vehicle. The service module would be ejected prior to deployment leaving the Pegasus package in an open position ready for expansion of the panels. Figure 20 shows a partial view of the Pegasus with one complete wing shown extended. The view of the camera is indicated in the lower portion of the illustration. A picture made by the camera during the process of panel unfolding is shown in Figure 21. The TV pictures were again transmitted to earth-based stations where they were recorded and did confirm the proper deployment of the wings of both the Pegasus satellites.

Figure 20 Saturn TV System with Pegasus Deployed (ASTR)

These selected examples have been given with the hope that they will provide foundation for additional ideas for use of television as an engineering tool. Those persons interested in additional information on any use described should contact James W. Wiggins - Technology Utilization Office, Mail Code MS-T, Marshall Space Flight Center, Huntsville, Alabama 35812, giving figure number and caption of use in question.

The author wishes to acknowledge the excellent work of the Photographic Branch of the Technical Services Office and the valuable assistance of the following in preparing this paper: Test Laboratory, Propulsion and Vehicle Engineering Laboratory and Astrionics Laboratory.

Figure 21 TV Picture of Pegasus Deployment (ASTR)

The Use of Fiber Optics in Optical Inspection and Monitoring

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ABSTRACT

The transport of optical images by means of flexible bundles of fibers permits visual inspection of otherwise inaccessible areas in tanks, piping valves and other enclosures. Both general purpose and specialized fiber optic devices have been built for this purpose. A general description of such devices is given along with a few examples of their application in optical inspection and monitoring.

GENERAL

The underlying principle of fiber optics is that light can be conducted by multiple internal reflections within a fiber of glass or other transparent medium. For maximum efficiency the fiber medium must have a high degree of transparency and this can be found in selected optical glasses and plastics. The fiber surface must also be as free as possible of any defects which could absorb or scatter light. For the latter reason most optical fibers are now coated with a thin, transparent coating whose refractive index is deliberately made somewhat lower than the fiber core. This coating serves both to protect the delicate interface at which the total internal reflections take place and to effectively separate the interfaces of neighboring fibers to prevent light from leaking from one fiber to the next.

By assembling a bundle of such fibers in a systematic array, i.e. with the identical fiber arrangement at both ends of the bundle, one obtains a device capable of transporting an image from one end face to the other. Such a device is often called a "fiberscope." Specifically, an image formed at the input end by

an objective lens, will be dissected into as many individual components as there are fibers. Each image component is transmitted independently of all other components even if the fibers are twisted, bent or intertwined. Upon reaching the output face all the image components are recomposed into a recognizable image where it may be viewed or photographed.

Since the size of the fiber limits the size of the smallest image component which can be resolved, it is appropriate to use as small a fiber size as possible. However, as the fiber size is reduced, it becomes increasingly more difficult to align the fibers in a precise geometrical array. To get around this dilemma, the "multifiber" technique was developed. By simultaneously fusing together a number of fibers into a single strand it is possible to make each fiber small while retaining a substantially larger "multifiber" building block which more readily can be assembled into a precise array. Typical flexible multifibers are illustrated in Figure 1 which is a photomicrograph of a small portion of a multifiber bundle. These multifibers con-

Figure 1 Photomicrograph of a portion of a bundle of multifibers.

tain 36 fibers in a 6 X 6 square array, where each fiber is about 10 microns in size. The spaces between the multifibers are filled with the organic cement used to hold them together at the ends of the bundle.

The quality of image transmitted by such a multifiber bundle is shown in Figure 2. This shows the image of an Air Force type

Figure 2 Image of resolving power target transmitted by a multifiber bundle.

resolving power target transmitted by a 6 foot long fiber bundle with a format of 8 X 10mm. The resolving power is found to be approximately 50 line pairs/mm for this high contrast target. Some image defects are discernable including a few broken or imperfectly transmitting fibers and some image distortions due to imperfect fiber alignment. Despite these imperfections there is a considerable amount of transmitted information. Thus, it is appropriate to consider such fiber bundles for use in optical inspection devices.

By virtue of the multifiber building block approach for the manufacture of fiber bundles it is possible to vary their cross-sectional area or "format" over wide limits from as little as 1mm in diameter to an inch or more. The length can also be varied over a considerable range from a few inches to as much as 15 feet, the latter representing the present limit of available tooling. Due to the unavoidable light absorption of even the best optical glasses there is a practical limit to the length of a fiber bundle somewhere around 25-30 feet where the transmission

is only about 8-10% and obtaining adequate illumination becomes increasingly more difficult. There are also fabrication problems which remain to be solved for such long bundles.

Figures 3 and 4 illustrate some of the range of presently available fiberscopes. Figure 3 shows a unit with a diameter of

Figure 3 Small fiberscope.

Figure 4 Large format fiberscope.

1/8 inch and a length of 24 inches, fitted with a tiny objective lens of about 2mm in focal length at the distal end and a 20X magnifier at the eyepiece end. The actual fiber bundle has a format of about 1mm square and contains about 40,000 fibers. Figure 4 shows a unit with a diameter of 1 1/2 inches and a length of 72 inches. The actual fiber bundle format in this case is 3/4 inches square and contains about 3,000,000 fibers.

One of the more common format sizes selected is equal to that of a standard 16mm camera aperture, namely 7.5 X 10.25mm. This allows for convenient coupling at 1:1 magnification to either a 16mm motion picture camera or a Vidicon TV camera. Such a fiber bundle is customarily housed in a rugged sheath capable of considerable physical abuse without damaging the fibers and thus lends itself to use under severe environmental conditions. An example of such a unit is shown in Figure 5. In this example the fiberscope is fitted with a standard 1-inch focal length ciné camera objective lens and a 10X magnifier for viewing the image.

Figure 5 Standard 8 X 10mm format fiberscope.

APPLICATIONS

With this background it is appropriate to consider some possible applications of fiberscopes in the areas of inspection and monitoring. Two of the most unusual applications are found in the Saturn S1-C or main Saturn booster stage being built for NASA by the Boeing Company. One of these is in the monitoring of the huge LOX tank of the S1-C booster. Two fiberscopes are used to photograph the behavior of the LOX during firing. One is equipped with an extreme-wide angle lens to view the tank when nearly full and the other with a medium angle lens to view

the tank when partially drained. Both are connected to 16mm motion picture cameras via 1:1 magnification relay lenses which separate to permit the cameras to be jetisioned for eventual recovery of the film they contain. The purpose of the system is to detect any unusual behavior of the LOX such as severe sloshing which could rupture the tank.

The fiberscopes make it possible to relay the image from the tank to the wall of the booster where the camera can be positioned for jetisioning. One of the two fiberscopes used in this system is shown in Figure 6. It is 9 feet long and approximately 8 X 10mm in format. It is shown equipped with the extreme wide angle objective lens and connected to a test camera by means of the relay lens.

Figure 6 Fiberscope used in Saturn Sl-C LOX tank.

The second system used in the Saturn Sl-C booster is shown in Figure 7. This system is designed to relay images of the rocket engines to a television camera for real-time monitoring of the engines to detect any malfunction. The unusual feature of this system is that the distal end of the fiberscope is divided into two branches, each of half the format area of the combined end. Each branch is fitted with its own wide angle objective and views one of the engines. Both formats are then combined at the output end and these are relayed with one relay lens to the television camera.

Figure 7 Fiberscope used to view Saturn S1-C engine nozzles.

In this case the fiberscopes permit the camera to be located safely behind a heat shield at some distance from the hot and noisy engines. The overall length of each branch of this fiberscope is 6 feet and the combined section is 2 feet long. The separate formats are 6 X 9mm and the combined format is, therefore, 9 X 12mm.

Another example of a monitoring system coupled to a television camera is shown in Figure 8. Unlike the previous examples in which the illumination was provided by external sources, this fiberscope is equipped with built-in illumination in the form of a separate fiber bundle which surrounds the objective lens and floods the subject with light. The light is provided by two projection lamps located in the housings on either side of the television camera. They are connected to the main fiberscope by auxiliary flexible fiber bundles commonly known as "light guides." In addition to internal lighting this fiberscope is also equipped with means for remote focussing of the objective lens. The length of the fiber bundle in this case is again 6 feet and the format is 8 X 10mm.

The application for which this system was designed is to view the casting of solid rocket propellant which is somewhat hazardous but requires visual monitoring. The fiberscope provides flexibility of viewing not readily achieved by the television camera alone. The video image is then relayed to the monitor located remotely at the control center.

Figure 8 Fiberscope with built-in illumination for monitoring casting of solid rocket propellant.

An example of a visual inspection device based on the fiberscope is shown in Figure 9. This might be considered a "classic" example in that it is the type of system probably envisioned by early workers in the field of fiber optics. It consists of a 12 1/2 foot long fiber bundle with an objective lens and right angle prism in the distal head and a 20X magnifier at the eyepiece end. A small lamp in the distal head provides the illumination while the transformer provides a variable voltage to the lamp. Such a fiberscope is especially useful for inspecting piping where the presence of coils or elbows do not permit a direct view or use of a conventional borescope.

A somewhat more elaborate version of an inspection fiberscope is illustrated in Figure 10. This device has the unusual feature that the first 15 inches of the housing can be articulated, i.e. bent to conform to almost any prescribed curve, by means of two control wheels located on the control box. This allows the system to scan the inside of a tank or enclosure after the system has been passed into it via a vent pipe or other channel. This system is also equipped with remote focussing of the objective lens. The fiber bundle is 12 1/2 feet long and has an 8 X 10 format.

This device was specifically designed for use in nuclear inspection and features a replaceable fiber bundle when the

Figure 9 Fiberscope for inspecting inside of piping.

original becomes too absorbing due to radiation browning. This is a severe problem in any fiber optics for use in any high radiation environment. The possibility of using silica fibers for this purpose is being investigated, but no results are available at this time.

These examples have been described primarily to indicate the type of fiber optic devices already in use in a variety of applications and to possibly suggest uses in the field of incipient failure detection in nuclear reactors. Within the limits of available length and flexibility it would seem that there are indeed possibilities for direct viewing with such fiber optic devices in this important field.

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Figure 10 Articulating fiberoptic for inspecting remote areas
 around nuclear reactors.

Discussion on Papers by Huggins and Siegmund

TENNEY (LASL): Mr. Siegmund, what is the smallest armature into which you can insert your optical instrument in pipes?

SIEGMUND: We make an instrument that can be inserted into a $\frac{1}{4}$ -inch-diameter hole; but, if it goes through a pipe, then it would have to be larger—somewhere around $\frac{5}{8}$ inch in diameter. Such a pipe could be negotiated by one of these small fiberscopes.

BALLARD (SANDIA): Can you say a word about the development of infrared fiber optics?

SIEGMUND: We have developed—but strictly in a laboratory sense—arsenic trisulfide fibers. These were capable of transmitting light in the wavelength range of 6 to 10 microns, with some very big holes in the spectrum where the water bands occurred. We have never built such fibers into an imaging-type device, however. We built only a light guide for transmitting infrared, and it is strictly experimental.

HIRSCH (ANL): What are the high temperature limits for these devices?

SIEGMUND: Right now the limits are imposed by the plastic that we use to cement the fibers together. I have not said anything about fused bundles. I would say that 250°F is probably the upper limit at the moment. If one only needs to introduce light (but not an image) into an area, it would be possible to fuse the fibers together at the tip and thus achieve a temperature limit of 800 or 900°F.

RICE (NASA): Mr. Siegmund, you talked about packing densities of 10^6 fibers/cm². When you increase the fiber density, you increase the resolution, but how does this affect the transmission of light?

SIEGMUND: Transmission is affected principally by length, not by cross section. The glass itself absorbs light, and there are reflection losses. Of course, there are spaces between fibers that do not transmit light, and there is cladding on each fiber which does not transmit light but merely insulates the fiber from its neighbors. For example, a 6-ft fiber bundle will transmit something like 25% of the light within the acceptance angle of the fibers, and, for a bundle 12.5 ft long, the transmission goes down to about 15%. Beyond that the transmission decreases exponentially. One could gain a little by using coarser fibers, but then that becomes a trade-off with resolution.

Analysis, Detection, and Control of Failures in Mechanical Equipment

BENO STERNLICHT
Mechanical Technology, Inc.

INTRODUCTION

"Mechanical Failures" cover a very broad spectrum of technologies and disciplines. These include fundamental analysis of failure mechanisms (such as wear, fatigue, corrosion, erosion and fretting), analysis of components (such as bearings, seals, gears and cams) and systems (such as turbomachinery, reciprocating machinery, transmissions, mechanisms, etc.), transducers for measuring internal and external parameters, diagnostic equipment and analysis (e.g., signature and pattern recognition), reliability, maintainability, etc. Several examples of failure mechanisms are illustrated (Fig. 1) and it has been demonstrated that these mechanisms can lead to catastrophic equipment failure (Fig. 2).

The preponderance of mechanical system failures can be traced to malfunction of a relatively few critical components such as bearings, gears, seals, piston rings, etc. MTI's approach to improve reliability and reduced maintenance of mechanical equipment is to investigate the principal modes of failure of these key machinery components and to apply the findings to improve design and to develop systems for detection of malfunctions and diagnosing the operating condition of the equipment. Such investigations are intended to provide the following types of information:

1. Identification and definition of the sequence of events and the cause and effect relationships which make up the failure process.
2. Correlation between the events occurring within the component during the inception of failure with the resulting effects as measured externally by sensors located at some distance from the components.
3. Establish relationships between the degree and rate of deterioration of the component and the performance of the machine of which it is a part.

WEAR

FRACTURE

FATIGUE

FRETTING

CORROSION

SPALLING

SCORING

TRANSFER

WELDING

PITTING

ABRASION

EROSION

Fig. 1 Mechanisms of Failure

Fig. 2 Component and System Failure

The effects of equipment failures are becoming increasingly more important and expensive. Because the machinery and the systems today are more complex, the maintenance and loss of product are very expensive and, in the case of military applications, often life is at stake.

The present machinery trend is to get out of the equipment more horsepower per pound of material. This is especially important for military applications, and it is accomplished by going to higher speeds, temperatures, compression ratios, BMEP's, and supercharge pressures. Table 1 gives an example of design trends between 1945 and 1965 in 30,000-pound diesel engines.

Table I Performance Increase for 30,000 Pound Weight Class Diesel Engines During Period 1945-1965

Period	Type	HP	BMEP	Piston Speed Ft./Min.	Exhaust T°F	Lb./HP	Supercharger Pressure Ratio	RPM
Prior to 1945	Stationary	800	110	1200	1000	37.5	Naturally Aspirated	720
	Locomotive	800	NONE IN USE					
1945 - 1950	Stationary	1000	130	1350	1050	30.0	1.3:1	800- 900
	Locomotive	1000	130	1350	1050	30.0	1.3:1	900
1950 - 1955	Stationary	1200	160	1500	1125	25.0	2.2:1	800 -900
	Locomotive	1200	160	1700	1125	25.0	2.2:1	850 -900
1955 - 1960	Stationary	1400	190	1700	1175	21.5	2.2:1	900
	Locomotive	1400	190	1700	1175	21.5	2.2:1	900
1960 - Present	Stationary	1600	210	1700	1300	19.0	2.5:1	1000
	Locomotive	1800	250	1900	1225	16.5	2.8:1	1100

Machinery insurance underwriters are faced today with claims which make them re-evaluate their business. One way of illustrating this point is to look at trends. In 1963 a \$50,000 deductible insurance was unusual but, as new ammonia plants came into being, deductibles rose to \$500,000 — and now \$1,000,000 deductible insurance is not uncommon. Evaluation of recent costs for turbine failures has shown an average cost of \$335,000 per failure and compressor failures have cost \$251,000 per accident (Ref. 1).

In Britain a survey was conducted which indicated that the cost of replacing worn machine parts was estimated at more than \$1.5 billion in 1965. In the United States this number was estimated in excess of \$10 billion.

Each manufacturer and user of equipment must ask himself the fundamental question: "How much does it cost to engineer new equipment and how much does it cost to 'fix' existing equipment?" The answer will be most revealing and, in most cases, will justify a failure prevention program.

Even though the failure reports are far from what an analyst would like, the data can be categorized by application and specific problem areas. Table II shows applications and components which are prone to wear failure. The most common machine elements that fail include bearings, gears, seals, piston rings and blading. All are working members transferring energy by sliding, rolling, and stress reversals. Thus, it is not surprising that many of the failures are those of wear (in the broad sense) and fatigue.

It is important to point out that the magnitude of the design parameters influencing failure is not necessarily the most important information required in assessing the performance of the equipment. In fact, information on rate of deterioration or propagation can often be considerably more informative about the degradation of the equipment. Rate measurements are especially important in assessing the time and frequency for performing maintenance on the equipment. In fact, a question can be raised of whether maintenance should be performed at all during the "design life" of the equipment. Fig. 3 shows a number of questions that must be answered by researchers, manufacturers, and users of equipment as we study the subject of "design life," maintenance and reliability.

A few examples will be cited here to illustrate the integrated program that we have undertaken in the study of "Mechanical Failures." This program integrates analysis, detection and control. Table III briefly outlines several of the studies included in this program. In order for the program to be of maximum value and have feedback to the designer, it must directly or indirectly correlate internal measurements with external measurements and analyses. This is schematically illustrated in Fig. 4.

ANALYSIS

"Failure" in the mathematical sense can be considered as "limit" of satisfactory design. This fundamental point is sometimes difficult for engineers to understand, therefore, some simple illustrations may be in order.

Hydrodynamic Lubrication

It is now well recognized that wear in mechanical components (bearings, gears, etc.) will not occur when hydrodynamic lubrication prevails. Therefore, it is important to determine the effects of various design parameters on the limits of hydrodynamic lubrication. Limiting analyses have to be performed in the following areas in order to answer:

1. At what speed will the rotor-bearing system in the hydrodynamic regime become unstable?
2. At what speed will the system no longer dissipate the generated frictional heat? Thus result in
 - a. lubricant breakdown
 - b. bearing material failure
 - c. bearing distortion and loss of load
3. At what speed and/or load will the asperities touch?

Table II Application and Specific Problem Areas in Machine Components

Application	SPECIFIC PROBLEM AREAS								
	Journal and Thrust Bearings	Rolling Contacts	Seals	Gears	Brakes and Clutches	Piston Rings	Electrical Contacts	Bearing Pivots	Wear Surfaces
Space Power Systems	X	X	X	-	-	-	-	X	-
Missiles	-	-	-	-	-	-	X	-	-
Rocket Engines	X	X	X	X	-	-	-	X	-
Satellites	X	X	-	X	-	-	X	-	X
Gyroscopes	X	X	-	-	-	-	X	-	-
Aircraft General	-	-	-	-	X	-	-	-	-
Gas Turbine Main Shaft	-	X	X	-	-	-	-	-	Dovetails and Interlocks
Accessory Equipment	X	X	X	-	-	-	-	-	-
Airframe	X	X	-	-	-	-	-	-	Control Surfaces
Laser Refrigeration Systems	X	-	-	-	-	X	-	-	-
Helicopters	X	X	-	X	-	-	-	-	Ball Joints
Surface Ships									
Low Maintenance Machinery	X	-	X	X	-	-	-	X	-
Water & Steam Lubricated Machinery	X	X	-	-	-	-	-	X	-
Hydrofoil Drives	X	X	X	X	-	-	-	-	-
Reactors									
Fission	-	-	-	-	-	-	-	-	Control Rods and Expansion Joints
Fusion	-	-	-	-	-	-	-	-	-
Nuclear Powered Turbo-Machinery	X	X	-	-	-	-	-	-	-
High Performance Weapons	-	X	-	-	-	-	-	X	Latches & Sliding Surfaces
Deep Submergence Submarines	X	-	X	-	-	-	-	-	-
Heavy Duty Construction Equipment and Off-The-Road Vehicles	X	X	X	X	X	X	-	-	X
Metal Working	-	-	-	-	-	-	-	-	Die Wear and Friction Loads

Fig. 3 Machinery Design Life

Table III Mechanical Failures

ANALYSIS	DETECTION	CONTROL
BASIC MECHANISMS	COMPONENT INSTRUMENTATION	IMPROVED DESIGN
WEAR FAILURES	STUDIES	REDUCE VULNERABILITY
VIBRATION AND FATIGUE		FAIL-SAFE DESIGN
CORROSION AND EROSION		MATERIALS
CREEP AND FRACTURE		
COMPONENT FAILURES	FAILURE TRANSDUCER	IMPROVED MAINTENANCE
FAILURE ANALYSIS	DEVELOPMENT	DIAGNOSTIC INSTRUMENTATION
INFORMATION ANALYSIS		CONTROL INSTRUMENTATION
		MODIFY OPERATING CONDITIONS
		IMPROVED MANUFACTURING
		QUALITY CONTROL
		TEST PROCEDURES

Fig. 4 Detection Systems

Many of these questions have been answered through such analyses as:

1. Bearing-rotor stability
2. Elastohydrodynamics and thermoelastohydrodynamics
3. Microelastohydrodynamics
4. Hydrodynamic, elastic and thermal analysis of bearings

Table IV indicates schematically which analysis must be performed in order to accurately describe the mechanism of operation that takes place in machine elements for various magnitudes of the design parameters. Such analyses (Fig. 5) were performed for mechanisms operating in elastohydrodynamic regime and limits of operation were established. Fig. 6 shows these results in dimensionless form (Ref. 2). It might be added that the agreement between theory and practice is very good.

Elastohydrodynamic and Boundary Lubrication

Temperature is a variable which is usually easy to measure and which is a critical parameter for failure in many mechanical elements and a good symptomatic indicator of the mode of operation of others.

Temperature is a critical parameter for lubricated surfaces which experience direct contact of either a continuous or occasional nature. This is because the protective or boundary lubricant films which produce low friction and wear are effective only up to a certain surface temperature limit (Ref. 3). Beyond that temperature, the films melt, or desorb, or redissolve in the bulk lubricant.

The critical surface temperature concept for lubrication has also been shown to correlate well with gear scoring data (Ref. 4). However, here

Table IV Applicability of Hydrodynamic, Elastic, Energy Considerations in Thin Film Lubrication

MAGNITUDE	DESIGN PARAMETERS						RESTRICTIONS
	L → H	L → H	L → H	L → H	L → H	L → H	
ANALYSIS	LOAD	SPEED	CONDUCTIVITY OF SOLID	STRAIN RATE	VISCOSITY PRESSURE INDEX	VISCOSITY TEMP INDEX	
ELASTICITY E	██████████			██████████	██████████	██████████	LOW SPEEDS ISOTHERMAL ISOVISCIOUS
HYDRODYNAMICS HD	██████████	██████████		██████████	██████████	██████████	LOW LOADS ISOTHERMAL ISOVISCIOUS
ELASTO-HYDRODYNAMICS EHD	██████████	██████████		██████████	██████████	██████████	ISOTHERMAL ISOVISCIOUS
EHD (VISC. PRESS. DEPENDENT)	██████████	██████████		██████████	██████████	██████████	ISOTHERMAL
THERMOELASTO-HYDRODYNAMICS ETHD	██████████	██████████		██████████	██████████	██████████	LOW STRAIN RATE
ETHD (ADIABATIC)	██████████	██████████		██████████	██████████	██████████	LOW CONDUCTIVE SOLIDS
ETHD (CONSIDER NON-NEWTONIAN AND VISCOELASTIC EFFECTS)	██████████	██████████	██████████	██████████	██████████	██████████	

L = LOW H = HIGH

the speeds are high so there is a sharp increase in the temperature of the surface as it passes through the loaded contact region. It is the maximum total surface temperature, including both the mean temperature and the flash or peak temperature within the contact region, which must be considered.

Temperature measurements may also be used as an indirect method of monitoring some other critical parameter which is more difficult to measure. In the case of tilting-pad thrust bearings, a theoretical analysis was performed to establish the relationships between operating conditions and the local film thickness and surface temperature over a bearing pad (Ref. 5). From the results of the analysis one can:

1. Specify the best location for sensors installed within the pad — that is, the points where the temperature variation with operating conditions, and hence with film thickness, will be greatest.
2. If the operating conditions are known, a measurement of the temperature in the groove ahead of the pad may be used as an indicator of the pad minimum film thickness. Fig. 7 shows some theoretical design data which could be used for this purpose.

Fig. 5a Failure Study of Rolling and Sliding Elements

Fig. 5b Flow Diagram of the Elastohydrodynamic Problem

Boundary Lubrication and Wear

Wear is probably the major cause of failure of mechanical components. Parts fail by wear because a critical positioning function between parts is lost, because the part is weakened to the point where it breaks, because there is an increase in friction to the point where functionality is hampered, or because close fitting clearances are enlarged resulting in leakage or other loss function. Table V classifies the wear processes that take place in mechanical components.

It is now recognized that there are several mechanisms of wear and that various parameters influence it. It is, therefore, not surprising that there are numerous theories, many of which are contradictory. Some agreement, however, appears to emerge. It is now recognized that solid solubility, crystalline structure, surface oxidation, coefficients of friction, sliding velocity and microhardness all affect wear. Three studies in this country are of particular interest (Ref. 6):

Fig. 6 Elastohydrodynamic and Microelastohydrodynamic Operating Design Limit

1. I.B.M. Recognizing the problems associated with wear of their computer mechanisms, I.B.M. undertook a concentrated study in 1952 called "zero wear design." This is strictly an empirical study with numerous experiments leading to empirical equations to be used by designers to insure acceptable limits of wear.
2. M.I.T. M.I.T. has established a controversial "metal compatibility" chart for selection of materials and the use of coefficients to predict measurable wear.
3. N.A.S.A. N.A.S.A. has concentrated on development of alloys of hexagonal crystalline structures for improved wear resistance.

Considerable effort is needed to pull the various theories together into a consistent integrated understanding that designers can use. It is not surprising, however, that this represents a monumental task because of the interaction of the many influencing parameters on wear.

Other Mechanisms of Failure

The subjects of fatigue, corrosion, erosion and cavitation are equally complex but they, like wear, have received considerably more

Table V Wear Processes in Mechanical Components

A CAUSE OF WEAR FAILURE	B PROCESS OF WEAR FAILURE	Journal Bearing	Thrust Bearing	Rolling Contact Bearing	Bushings	Pivots	Seals	Brakes	Clutches	Piston Rings	Electrical Contacts	Wear Plates	Gears		C RESULT OF WEAR
Dirt	Material Removal			X	X	X	X			X					Seizure
Corrosion	Surface Damage 1.	X	X		X		X				X	X	X		Improper Position
Misalign- ment	Thermal Instability 2.	X	X	X				X		X			X		Undesired Friction
Faulty Design	Fatigue	X	X	X											Breakage
Improper Materials	Material Distortion or Breakage			X			X						X		Noise
Over- Heating	Material Deterioration										X				Diminished Performance
Lubricant Interrup- tion	Frictional Instability 3.				X			X	X						
Improper Lubricant															

1. Damage to the surface from relative motion while in contact.

2. Heat generated fails lubricant, etc.

3. Friction to high or low, variable friction, vibrations.

Fig. 7 Calculated Relationships Among Operating Variables for a Tilting-Pad Thrust Bearing

attention in the last few years. Therefore, we are beginning to understand the interaction between the various parameters and can come up with better criteria for design. To illustrate this growth in interest, a study was performed which indicates that the open literature on fatigue has a 15% average annual increase. Approximately 6,000 papers were listed by ASTM since 1950 (Ref. 7).

A multimillion-dollar-per-year "Coupling Program" on corrosion is presently sponsored by ARPA which couples technical talents from industry, government and universities. This is in recognition of the corrosion failure problems that occur in equipment. In addition, it is appreciated that a coordinated program, in which various groups participate, can be of greatest value. One such program recently undertaken is the ONR "Mechanical Failure" Program. Several government agencies are sponsoring this program.

"Limits" of satisfactory design apply to all components and systems. Aerodynamic components (compressors, turbines, etc.) may have the same or different design criteria than mechanical components (bearings,

gears, seals, etc.). For instance, in the case of compressors, the following "limit" analyses are in order:

1. When is the stress too high so that the impeller or blading rupture?
2. When does the centrifugal and thermal growth interfere with other members of the machinery?
3. When does the machine go into surge?
4. Etc.

Vibration and Noise of Systems

Analysis of the "Limits" of satisfactory design of a system as opposed to components, in general, is more complex because the various interactions have to be considered. Unacceptable performance may result from excessive vibration or noise. One must know the magnitude of the forces and the attenuation paths in order to analytically predict vibration and noise of a system.

MTI has a program underway sponsored by the U.S. Army to analyze gear noise. A principal purpose of this program is to develop analytical tools which can be used to evaluate the potential noise producing characteristics of gearboxes, with particular reference to helicopter transmissions. The normal operation of power transmission gearing results in the generation of noise and vibration. The noise which is generated occurs principally at the tooth meshing frequencies and their harmonics. It is caused by the fact that the meshing action of the gear teeth produces errors, or deviations from a perfectly uniform angular rotation of the mating gears. The vibration and noise which a set of gears makes depends on the profiles of the teeth and on the applied load and speed. Since the principal modes of gear failure are accompanied, and usually preceded, by some change in the detailed profile of the teeth, we have a promising situation for evaluating the condition of the gears and detecting advance signs of failure through measurement and analysis of noise or vibration.

The gearbox noise analysis consists of two parts: first, a method and procedure for computing the error, or deviation from uniform rotational motion, of the meshing gears; and second, a procedure for computing the dynamic forces potential noise producing power of the system of gears and shafts which makes up a specific gear transmission.

Theoretical analyses are in good agreement with practice (Ref. 8). Fig. 8 shows "signatures" of three helicopter transmissions. Each shows some patterns with somewhat different magnitudes of noise.

DETECTION

As illustrated in Figs. 1 and 2, initiation of failure in one component may result in catastrophic failure of the entire equipment. Thus, as in the process of fatigue, we have "initiation" and "propagation" of failure. Initiation may be in one component which may propagate to another, and the observed major damage may then be observed in yet another part of the system. Therefore, nearly every failure leads to numerous concomitant failures in the chain reaction that follows.

Fig. 8 Sound Pressure Level Versus Frequency for Helicopter Transmissions

For this reason it is important to select the right detection equipment and to make sure that it is accurate, reliable and installed in the proper location, in order to give early indication of incipient failure. Three direct benefits will result from its use:

1. avoid catastrophic failure
2. generate "feedback" information for improved design and/or modified operating procedures
3. evolve more rational criteria for maintenance action.

There are numerous detectors and readout systems that are used to measure mechanical parameters. In this paper only a few of those which MTI has concentrated on will be described. In general, they fall into "noncontact" instrumentation (Ref. 9). Many detector probes are possible; they include capacitive, inductive, optical, etc. Fig. 9 illustrates just a few of these. Probes of various sizes and shapes have been developed in each of the capacitive, inductive and optical areas (Fig. 10). The applications of these probes are many and several are illustrated in Fig. 11. Fig. 12 shows several systems that have been developed and which are presently commercially available. Simple and reliable equipment for measuring motions of 0.0001 inch from a stand-off distance of one inch "Foci tronic" and angle-sensor for measuring motion of less than one second of arc ("Refractosyn") are now available.

Several other transducers that have been developed at MTI are worthy of mention. The first of these is a wear detector.

Fig. 9 Representative Types of Non-Contact Instrumentation Probes
 (From left to right: A capacitance probe for use with the MTI Captronic Sensor; An inductance probe for the MTI Indtronic Sensor; A Photocon capacitance probe with integral in-place calibration check; An MTI Fotonic Sensor fiber optic probe.)

Wear may be measured directly — for example, by measuring the change in relative position between mating parts using displacement sensors. Or a sensor may be incorporated into the wearing surface so that some measurable property of the sensor will change in a predictable fashion as wear takes place. One such concept is the Electrical-Resistance Wear Detector illustrated schematically in Fig. 13. The cylindrical sensor may be 0.1 inch in diameter or less and is installed in the wearing surface flush with the surface. Fig. 14 illustrates a typical wear versus resistance characteristic obtained with such a sensor. The sensitivity, slope of the curve, and total wear range are variable depending on the height of the wear resistor section and choice of the conducting film material.

Wear may be monitored by indirect measurement if the effects of wear on other system parameters are understood. For example, wear of either part of a journal bearing will increase the bearing clearance. If the bearing is supplied with lubricant under pressures as most are, the lubricant supply rate will increase as wear increases the clearance. Theoretical design data are available to establish the necessary relationship between clearance and lubricant flow rate. Changes in clearances or the fit of mating parts usually results in changes in

Fig. 10 Noncontacting Detectors, Sizes and Shapes — Capacitive, Inductive and Optical

the vibration and noise generated during operation. If the nature of these changes is recognized, then external noise and vibration measurements can be correlated with component wear.

Another example of a detector is the miniature strain gage and printed circuit to measure local strains. Gages of 0.015-inch length have been effectively used to measure stress concentration in fillets (Ref. 10). Fig. 15 shows hypoid gear tooth models strain gaged for tests. Fig. 16 shows printed wiring strain gage installation. Fig. 17 shows loading fixture for applying point loads in a 3 X 11 matrix. Geometric similitude was employed in this study by using aluminum in place of steel in order to increase strain. This technique proved to be extremely successful and can effectively be used in many other applications.

One of the most significant recent instrumentation developments is the laminated capacitance pressure transducer. This transducer is expected to be as valuable as the strain gage. It measures local pressures without interrupting the flow. Fig. 18 shows a sketch of the transducer and its characteristics. Because of its small size, it is particularly well suited for pressure and pressure gradient measurements on air foils, hydrofoils, turbine vanes, propellers, and similar components. Both static and dynamic pressures can be measured with a single transducer.

Fig. 11 Examples of Applications for Noncontacting Instrumentation

Fig. 12 Capacitive, Inductive and Optical Readout Equipment

Fig. 13 Illustration of Electrical Resistance Wear Sensor Before and After Wrapping on Center Support Rod

Fig. 14 Characteristic Wear Versus Resistance Relationship for Electrical Resistance Wear Sensor

Fig. 15 Hypoid Gear Tooth Models Strain Gaged for Tests

Features of the new transducer system are:

1. Transducers and lead wires are very thin and can be applied in such a way that flow is not disturbed.
2. Transducers are very light and can withstand high acceleration in the plane of the transducer.
3. Resonant frequency is high.
4. Resolution is high.
5. Size is small, permitting installation of many transducers and wiring on a small area.
6. Shape is not limited to circular. Any shape desired can be used.
7. Design is flexible and can be tailored to specific applications.

The principle has been reduced to practice successfully and patents have been applied for. We are now in a position to provide the equipment and consult in its application to a specific problem.

The use of ultrasonics is also finding many applications to the field of failure detection and diagnosis. One example of this is the ultra-

Fig. 16 Top View, Showing Printed Wiring Strain Gage Installation. Magnification 2.3X (Viewed at Slight Angle With Vertical)

Fig. 17 Side View of Loading Fixture for Hypoid Gear Tooth Model, With Specimen Loaded on Concave Side

sonic converter or detector which senses the ultrasonic noise generated by rubbing surfaces without responding to lower frequency vibration in the audio range. As asperities touch, they generate noise of ultrasonic frequencies. This instrument may be very valuable as a means of monitoring the condition and operation of components which normally operate with some rubbing; such as seals, rolling element bearings, etc. Run-in, lift-off and wear could be investigated with the aid of this instrument. Another example is the application of ultrasonics to detection of fatigue cracks (Ref. 11). Both reflection and through-transmission techniques have been successfully used for detecting crack initiation and crack propagation prior to fracture. This technique has the advantage of being able to detect small fatigue cracks nondestructively during the course of a fatigue test.

Vibration, noise, radiation, oil and exhaust detectors have all been used to monitor and detect malfunctions of equipment. These, in general, are external measurements which can be correlated with internal measurements and parameters of design. Table VI shows the components, design parameters and measurements that are important in failure analysis and control. There is no single universal detector that will measure all the design parameters of interest. A minimum number of detectors located in carefully selected places can provide considerable information for analysis, design, control and measurement of degradation.

Fig. 18 Thin Laminated Capacitive Pressure Transducer (Patent Applied for)

CONTROL

An understanding of the interaction of the various parameters on mechanical failure is essential in order that failure can be controlled. The detection instrumentation with a logic circuit can effectively be used to monitor and control operation of equipment.

To a significant extent, mechanical failures can be traced to a high proportion of failures occurring in a few critical or "failure sensitive" components; bearings, gears, seals, rings, pistons, etc. These components contain certain optimum locations where incipient failure may be monitored in any machine. Research to a very significant extent must provide a basic understanding of the failure process in these critical components; the effect of variables on the process, how deterioration of these components affect system performance, and importantly, what type of instrumentation will detect impending failure.

Table VI Design Parameters and Measurements in Machine Components

COMPONENTS	PARAMETERS	MEASUREMENTS	
		INTERNAL	EXTERNAL
BEARINGS	LOAD	WEAR	ACCELERATION
GEARS	SPEED	FRETTING	VELOCITY
SEALS	TEMPERATURE	SPALLING	DISPLACEMENT
PIVOTS	MATERIALS	SCORING	OIL INSPECTION
BRAKES	TIME	TRANSFER	ACOUSTIC
CLUTCHES	ENVIRONMENT	WELDING	TEMPERATURE
RINGS, CYLINDERS		PITTING	OIL FLOW
CONTACTS		ABRASION	FUEL CONSUMPTION
WAYS, GUIDES		FATIGUE	TORQUE
CAMS		BRITTLE FRACTURE	
		CREEP	
		CORROSION-EROSION	

The tools are at hand to develop diagnostic systems which will detect failure and/or deterioration in any given piece of machinery. These systems will comprise an integration of various internal and external measurements.

To design the optimum monitoring and/or control system, it is necessary for the machinery manufacturer to correlate data on the way in which components fail under operating conditions, including failure time and overload tolerances. Fig. 4 shows the correlation of design information with respect to detection systems. Table III shows the relationship existing between analysis, detection, and control. The acquisition and correlation of complete data relating to failures would result in more adequate failure detection equipment. It would also lead to improved design of critical components, superior equipment, better maintenance programs, and eventually substantially reduced failure histories.

Failure analysts are aware that it is rare for failure to occur due to a single cause. In most cases, a complex of interacting circumstances conspire to cause the final catastrophe. Table VI illustrates this fact. It follows that complete answers to the following questions are a fundamental necessity in the development of an adequate monitoring system:

1. What is the sequence of events which lead to failure?
2. What effects would the operating variables have on the failure process?
3. What was the effect of design variables?
4. Which measurements are best able to sense impending failure and what degree of accuracy is required?
5. What steps can be taken to arrest the failure process once it is started?
6. How do impending failures and proceeding failures affect overall machine operation?

Broad assessment of the potential problem with the equipment can serve a very useful function to the manufacturer, contractor, and user of the equipment. Some examples will be cited to illustrate this point of view.

Vibration suppression is a vitally important aspect of the design of any high-speed or high-power geared transmission system. Geared systems are subject to torsional vibration of the whole system, bending vibrations of drive shafts, plate vibrations of gears, axial motions of certain subassemblies and various combinations of these vibration modes.

Three principal factors promote vibration problems in powered systems:

1. High power flow
2. High speed
3. System complexity

In Table VII these factors are evaluated in terms of their potential hazardous effect on system or operator.

It is difficult and sometimes costly to determine at the design stage whether a system will encounter hazardous vibrations during operation. Within limits, however, Table VII may be used as a reasonably effective guide, to ascertain preliminarily, the extent to which the system will probably be adversely affected by dynamic influence. Detailed analyses should only be performed where the hazard is high.

A reciprocating engine prime mover in a system with its inherent unbalance harmonics of torque entering the rotational drive will worsen any situation. Alternatively, the absence of such a vibration source is no indication that troublesome vibrations will not occur. These may still result from either a speed or power high enough so as to be capable of exciting heavy vibrations leading to failure without any other stimulus. Table VIII shows certain groupings of environmental factors which may be used to determine whether or not a problem is likely to exist. These charts are essentially qualitative, but when used together with discretion they can be of value in evaluating the need for more detailed analysis.

Table VII Degree of Potential Hazard

Factor	Degree of Hazard			
	None	Slight	Quite Probable	Almost Certain
Horsepower				
1	x			
100		x		
1000			x	
10000				x
Pitchline Velocity				
1500 ft/min	x			
4000		x		
10000			x	
25000				x
Arrangement				
Single mesh		x		
Multiple reduction			x	
Multiple inputs or outputs				x
Flexibility and Accuracy				
High-accuracy teeth, light masses, flexible shafts		x		
Medium-accuracy teeth, heavy masses, stiff shafts				x

An understanding of the interaction of various parameters can be used to advantage in controlling the behavior of equipment. An example of rotor-bearing response will illustrate this point.

The amplitude of vibration of rotor bearing systems depends on spring and damping coefficients of bearings, number of bearings, their location, rotor mass and inertia distribution, and operating speed relative to critical speed. Amplitude of vibration may be changed by introduction of additional bearings or increasing damping as the rotor is accelerated or decelerated through the critical speed. In fact, a machine can be designed with controlled amplitude of vibration. The two patents, 2,961,277 — Functional Bearings, and 3,124,395 — Method and Apparatus for Bypassing Critical Speeds of Rotors, cover this subject. Using sensors to detect amplitude of vibration or speed, bearing damping and/or number of active bearings can be changed so as to limit the amplitude of vibration. With the aid of rotor-bearing dynamic analysis, the controls can be automatically programmed to limit the amplitude of vibration. Also, with the aid of instrumentation for detection of amplitude of vibration, the controls can be actuated. This example illustrates the point that analysis and monitoring instru-

Table VIII Assessment of the Extent of Problem

(a)

(b)

(c)

mentation can be effectively used to control the performance of equipment.

CONCLUSIONS

There are ways and means when employed in a systematic program to materially reduce the incidence of mechanical failure. Such a program must seek to answer several basic questions such as:

1. Has the proper equipment been selected to do the job in question?
2. Is the equipment sufficiently reliable and free from mechanical problems which can, over a period of time, result in down time and marginal operation?
3. How can we establish by means of test that performance and general mechanical integrity are sound?
4. What can be done in terms of monitoring to detect incipient problems?

5. How can we insure long-term reliable operation by preventative maintenance and inspection?

For an overall failure program to be most valuable, studies in analysis detection and control should be performed. These studies should be both fundamental and applied and they should provide feedback to designers in order to improve reliability of new equipment (Fig. 4).

RECOMMENDATIONS

It is recommended that the various studies shown in Table III be pursued in order to improve the life of equipment and to reduce maintenance.

It is further recommended that the manufacturer and user follow the five items listed below in order to insure reliable operation of equipment.

1. Employment of properly detailed purchase specification

This will insure that equipment being purchased for a given application has, in fact, been designed for realistic operating conditions in all respects, and manufactured with proper quality control and inspection criteria being applied. This item gives preliminary insurance that the machine is adequate for the function intended, and that it will integrate into the system without catastrophe.

2. Design audit

The equipment should receive a design audit to check performance capability, to establish mechanical integrity, and to determine the adequacy of critical items.

3. Proper engineering analysis of all of the machinery to an interconnected system.

4. Incorporation of monitoring instrumentation

This instrumentation is designed to detect the functioning of the critical elements and sense incipient failure; i.e., before it progresses to become the initiator of a catastrophic failure involving a component, a whole machine, or even an entire system.

5. Comprehensive preventative maintenance programs to encompass all factors requiring inspection and maintenance.

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Mechanical Signature Analysis : A New Tool for Product Assurance and Early Fault Detection

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Abstract

Sound and vibration signals have been useful for many years in judging the internal condition of machinery and structures. An automobile mechanic listens to an engine for internal defects. A machinist checks a grinding wheel for cracks by rapping it and listening for a clear tone or a dull thud.

At General Electric's Research and Development Center, the interpretation of sound and vibration signals is being developed from an art into a scientific technique. This technique, Mechanical Signature Analysis, uses external measurement of sound and vibration signals to diagnose internal condition or malfunctions and to detect incipient failure.

Mechanical Signature Analysis has great potential as a unique tool for increasing the reliability and maintainability of a wide range of components, assemblies, and systems.

Introduction

With the rapidly increasing demands for product reliability, it has become apparent that conventional screening, check-out and fault detection methods must be re-examined and improved.

It is well recognized today that to achieve high and predictable reliability for a system, it is not sufficient just to build it from high quality components. Many defects may be initiated first during assembly or during qualification tests. Such defects, which are not connected to component quality, reduce the overall system reliability. It is essential, therefore, that effective methods be available for checkout of a completely assembled product from the outside, before it is finally accepted.

But high reliability of a product may only solve part of the problem. What we really desire is high availability, i. e., short total down-time for maintenance or repair as compared to effective operating time. Most equipment needs maintenance, and the key to high availability, once the reliability (mean time between failures) is fixed, is to know early enough when preventive maintenance or replacement should be done. Therefore, a second essential need calls for methods which can detect an internal malfunction early, during actual operation, so that preventive maintenance can be scheduled before critical failure occurs.

These are two general areas, final product checkout and early fault detection where effective diagnostic methods can significantly increase total product availability. In both cases the central problem is to evaluate internal conditions through external measurements, without disturbing the normal operation of the tested equipment. The need for such techniques applies to a vast variety of products, including all kinds of machinery, static structures, and electronic assemblies.

This paper describes one approach to the problem which has been taken by the Research and Development Center of the General Electric Company in developing fundamental diagnostic tools for detection of internal malfunctions and defects.

Sound as Information Carrier

Obviously, external detection of internal defects presents new problems compared to conventional component testing and inspection. This is particularly true for mechanical defects. Internal cracks, loose bolts, worn bearings, etc., are no longer open for visual inspection or direct measurement. Therefore an information carrier is needed, to transmit the internal information to the external evaluator.

Utilizing different such information carriers, a new generation of evaluation tools has recently begun to evolve. It includes the use of many new techniques such as

New X-ray methods for detection of cracks, flaws in structures,

RF-noise detection of poor electrical connections,

Ultrasonic inspection for detection of materials fatigue, cracks,

Infrared radiation inspection, for detection of abnormal thermal conditions.

These methods have usually been applied in the screening of high reliability items, and have added further sophistication to earlier accepted regular quality control procedures. Some are now developed to practical usefulness in particular applications. But their limitation is that each method is rather specialized. It detects a certain kind of defect but is not general enough to cover the wider range of possible faults that can be encountered in a complicated system.

Here sound and vibration methods present additional potential for development into a wide range detection tool for internal mechanical defects. Sound and vibration are excellent information carriers in mechanical structures. And the fundamental mechanical events which occur in operating machinery, such as rolling, sliding, impacts, etc., all produce sound of some kind, which tells the initiated listener much about the internal condition.

Listening is really the oldest of all evaluation methods, basically because it does not require any instrumentation at all. A mechanic listens to an engine to determine its condition; or he might hold a screwdriver against a bearing housing to detect a malfunction in that particular bearing. Even entirely passive structures can be mechanically excited to produce a sound signature which is characteristic for its internal condition. Loose particles in a closed box or cracks in ceramics or glass, for example, are usually detectable that way.

Direct listening without special instrumentation is extremely useful for quality control or early malfunction detection in many cases. Very often changes in sound signatures precede the actual performance deterioration of machinery, which can be detected through other means. The sound from an automobile engine for example usually tells about a faulty valve long before it is detectable through oil pressure and power output changes.

However, this kind of intuitive evaluation is more an art than a science. It requires a certain skill, which often is very high, on the part of the listener and the result is not amenable to quantitative definition. To make acoustic signature evaluation really valuable as a modern screening or failure prediction tool, much has to be added in terms of problem definition and mechanized evaluation. It is necessary to develop new sensitive and selective methods which are able to detect subtle changes of certain kinds in a sound or vibration signal, and it is also necessary to understand the mechanics of the equipment under study in order to know what changes to look for.

Development of Mechanical

Signature Analysis

A Mechanical Signature Analysis program, directed toward the interpretation of external sound and vibration signals to determine the internal condition of machinery and structures, was started in 1964 by General Electric's Research and Development Center.

The general scope of this program is to explore all areas where sound and vibration techniques may provide sensitive, selective, and economical diagnostic tools for the industry.

Included within this very wide scope are all types of sound producing objects from small components like relays and bearings to large assemblies and systems like locomotives and spacecrafts.

The methods which are being used also vary within a wide range, as will be demonstrated later in this paper. It is not a question of developing any

particular sound analysis technique or instrumentation, but rather to use all that modern electronics has made available, including digital computer techniques.

Problem Approach

The uniqueness of Mechanical Signature Analysis lies not so much in the mere use of sound to determine internal conditions. Sound is being studied for this purpose elsewhere. But in order to be successful it is necessary to do more than just listen or measure obvious sound quantities.

The problem is usually to detect an internal defect while it is still very small, and its characteristic sound or its "signature" hidden by sound from many other sources. Therefore a selective technique must be developed in each application to extract from a complex sound or vibration signal some characteristic feature, a discriminant, which is highly correlated to the internal condition under study. This means a technique with the ability to sort out all kinds of "false alarm," including high background levels, and thereby reliably reveal a very small signature change.

It becomes the central problem to find this most effective discriminant in each application. There are several approaches possible:

One is entirely statistical. If a sufficiently large sample of signatures from good and malfunctioning units is available, it is possible to have a computer evaluate a large number of discriminants, and through a correlation program select the most effective. Considerable work is presently being done in this direction, but is not within the scope of the program we are discussing here. The success of course, depends entirely upon a good statistical sample and in many cases such a sample is not available or unreasonably expensive to obtain.

A second approach, which forms the basis of the Mechanical Signature Analysis program, is based on understanding of the process through which a particular signature is generated and transmitted through a structure. This usually requires an initial thorough study of the mechanics of the equipment to be investigated, so that an analytical model for normal and abnormal signatures can be established. Once this is done, an analysis method with a good chance of success can be selected and then optimized experimentally.

This approach offers many outstanding advantages. Only a small number of tests is required. Essential also is that if a partly effective method has been found, the theoretical knowledge enables one to predict how it must be changed to improve.

Experience has shown that key requirements for success are:

1. Understanding of the mechanics of the equipment generating the signals, its failure modes and failure mechanisms.
2. Understanding of the process by which normal and malfunction signals are generated and propagated from source to sensor.

3. Correct selection of sensor, i. e., microphone, accelerometer, or other transducer responsive to the right frequencies.
4. Based on 1 and 2 above, correct selection of a technique which extracts the desired malfunction signature from the raw vibration data.

Areas of Application - Present State of the Art

Mechanical Signature Analysis techniques have presently been applied to a number of problems in the areas of product assurance and preventive maintenance. Considering the types of systems which have been involved, the applications generally fall within one of two categories, active or passive systems.

An active system is a machine or a structure which generates its own vibration or sound signal, i. e., a motor or a pump. The problem generally concerns early detection of internal deterioration in order to prevent critical failure.

A passive system is a structure which is quiet in itself, i. e., a package of electronics, a structural frame. It can be vibrated or otherwise excited to generate a mechanical signature which then reveals the internal conditions. The problem is generally to detect internal cracks, loose fits, loose particles, etc. without disassembly.

In several cases feasibility of a certain method has been proven and practical detection systems are under development. In a couple of applications the development has been completed and systems are in practical use as screening methods for product assurance.

However, the general technology of MSA is in a state of early development and several fundamental problems on signature generation and propagation still need to be studied. It can be expected that many new systems useful in quality assurance and reliability improvement will evolve during the next few years.

In the following sections some groups of applications where Mechanical Signature Analysis have proven to be useful will be discussed. One particular application, a malfunction detection system for ball bearings, will be described in some detail.

Application to Active System

Active systems which have been investigated through Mechanical Signature Analysis include many kinds of machinery and components such as combustion engines, gear transmissions, pumps, fluid valves, electric power switches, ball bearings. They all generate their own sound signature during operation, so the fault detection device becomes passive, listening.

For discussion of the active systems it is suitable to separate them into three groups,

Cyclic Machinery (engines, transmissions)

Flow Noise Generators (pumps, boilers)

Single Transient Generators (valves, switches)

Different characteristics in the signature generation process has resulted in different analysis methods for the three groups.

Cyclic Machinery - Summation Analysis

A large group of machinery and components have cyclic operation which produces a sound or vibration pattern that repeats with a certain time interval. Three areas where mechanical signature analysis has been applied are

1. Detection of internal malfunctions in Army tank engines through external vibration measurement during operation.
2. Detection of wear in high precision gear trains through sound measurement during operation.
3. Detection of early fatigue damage in ball or roller bearings through vibration measurement during operation.

Sound and vibration data recorded during the operation of engines, bearings, and gear trains is very complex and often contain a high degree of background signals. Therefore, if a piston gets damaged in the engine or a fatigue crack appears in a bearing race, something changes in the signal, but the high background level makes it impossible to detect the change by simply measuring the overall level of the signal or some other obvious characteristic. It is necessary to develop a selective method which sorts out the background and clearly displays the part of the signal which is generated by the events under study.

In cyclic machinery the operational events repeat with known time intervals. For example, in a four stroke combustion engine the repetition period is two revolutions of the crankshaft. It would be logical then to use a detection method which enhances all features in the raw vibration signal that repeat every two revolutions, and discards everything else.

To accomplish this, a digital summation technique was developed, and found to actually be the most effective tool available for detecting many internal malfunctions in cyclic machinery.

An example of its function is shown in Figures 2 and 3, which display summed vibration signatures over a 720° engine cycle, obtained through summation analysis of two Army tank engines (Figure 1) undergoing maintenance at the Frankford Arsenal. Figure 2 shows the signature from a normal engine, Figure 3 the signature from an engine with a loose wrist pin. The additional peaks in Figure 3 correspond to impacts occurring at certain crankshaft positions.

Figure 1 Army Tank Engine
(V-12 Combustion Engine)

0° 360° 720°

Figure 2 Good Engine. Vibration Signature
for a 720° Engine Cycle, Obtained
Through Summation Analysis

0° 360° 720°

Figure 3 Engine with Loose Wrist Pin.
Vibration Signature for a 720°
Engine Cycle Obtained Through
Summation Analysis

The power of this method is that a particular peak, through its location in the summed vibration curve, and previous knowledge about the engines kinematics, can be related to a certain event in the engine. The technique is therefore highly selective; it is even possible in some cases to predict how a certain malfunction will show up in the analysis.

In addition to the loose wrist pin, sticking valves and damaged piston rings could be detected without disassembly through summation analysis. Some of these malfunctions could not be detected through any other method.

Flow Noise Generators - High Speed Spectral Analysis

Fluid in a dynamic state produces sound. This state can be motion of fluid in a pump, boiling in a reactor, motion through a chemical process, etc. Since the sound travels easily through the fluid, it can propagate information about a condition at a point where no measurement can be made to a point where it can be picked up by an acoustic sensor.

One application concerns early detection of cavitation in pumps. Developmental work is presently being done in this area. If an acoustic signal can reveal the initiation of cavitation before it has grown to a point where the risk for critical failure is high, it can be used to regulate the pump speed so that the pump always works efficiently but without cavitation. This increases the reliability of the system.

A very effective display format for this type of signal is obtained through High Speed Spectral Analysis. A unique analyzer of this type was developed at the GE Research and Development Center initially for the study of biological sounds. This analyzer operates in real time and makes a spectral analysis of the studied signal while maintaining a very high time resolution, so that a three dimensional frequency-amplitude-time display is obtained. This display makes it possible to study how the spectral content of a signal varies with time.

One brief example of this technique is shown in Figures 4 and 5. The problem concerns early detection of cavitation in the oxidizer turbo-pump for the H-1 Saturn rocket engine (Figure 4). Fluid borne sound was measured during pump operation. Figure 5 shows the spectral analysis from one run. Energy in the signal is displayed as dark areas in the figure. The vertical lines are cavitation transients. The horizontal line at 1100 cps is a blade passing frequency, and shows that the pump was run at constant speed. The sloping lines are related to fluid resonances. They change frequency when the bubble content in the fluid changes and are thereby related to cavitation.

A more extensive discussion of the cavitation signature study is given in lit. ref. 4 by W.B. Giles, F.O. Rathbun, and D.E. Wood.

Single Transient Generators

A third group of active systems consists of relays, electric switches, valves, etc. which have a very short work cycle and usually generates a set of impact transients while operating.

Figure 4 Instrumented H-1 Rocket Fuel Pump

Figure 5 Time-Frequency-Intensity Display of Fluid Borne Sound in Operating Pump

Mechanical Signature Analysis has been applied to several problems in this area. For example, it has been shown that wear in certain types of valves is detectable through evaluation of sound earlier than through any other non-destructive method.

Application to Passive Systems

Many passive systems, without any mechanically operating parts, fail for basically mechanical reasons. The initiating factor is normally an environmental exposure of some kind in connection with an initial mechanical defect.

Numerous examples can be given:

An electronic spacecraft assembly contains a small loose object, a piece of wire, a solder ball, etc. The object is not detected during the quality control procedure because it is trapped somewhere behind an electronic component and does not interfere with the operation of the electronics. When the assembly in flight is exposed to zero-gravity conditions, the loose object begins to move around and finally shorts an electrical circuit or a relay contact.

Another electronic assembly is subjected to a final acceptance test including environmental exposure. The assembly is sealed, checked for electronic performance, exposed to a high level vibration program and then checked again. This final test shows that no electronic failure has resulted from the vibrations and the assembly passes. What the test does not reveal, however, is that the vibration exposure has damaged the internal mechanical structure, maybe seriously reduced the strength of a bracket through a fatigue crack. There is no conventional method of detection of this mechanical defect since the box is already sealed, and opening would necessitate requalification. When the assembly is subjected to further vibrations later in its life, the bracket breaks catastrophically and causes failure.

This type of failure can seriously limit the total system reliability in many applications. The damage causing environment may be something as easily encountered as rough handling or ground transportation of equipment. It is therefore highly desirable to develop an effective checkout method which can evaluate internal structural condition from the outside, after the final performance tests. X-ray inspection is here a step in the right direction, but becomes of limited value as the complexity of the structure under study gets high. Acoustic methods again offer the basic advantage that sound and vibration information easily propagates through structures.

Impedance Methods

The most general method for external evaluation of structural condition appears to be measurement of mechanical impedance. The structure under study is mounted to a shaker through a special transducer, an impedance head, which measures both the vibration force that the shaker imposes on the structure and the resulting vibration level (Figure 6).

Figure 6 Measurement of Mechanical Impedance

For sinusoidal vibration excitation, the mechanical impedance Z is then defined as the ratio between vibration force F and the resulting vibration velocity V .

$$Z = \frac{F}{V}$$

or, since the vibration acceleration A is the normally measured quantity

$$Z = \frac{F}{A} \cdot j 2\pi f$$

Z is a complex function, usually depending on the forcing frequency. It can be considered as the mechanical equivalent to electrical impedance.

The usefulness of mechanical impedance comes from the fact that it, measured at only one point, gives information about the dynamic properties of a whole structure. All structural changes which involve stiffness, masses, or damping, such as cracks or loose fits show up in the impedance function. Another advantage is the fact that impedance measurement often can be incorporated as part of a regular vibration test, even if only a random vibration spectrum is available (lit. ref. 1).

The direct interpretation of impedance functions in terms of structural defects in a complicated assembly is very complicated and needs much further development. But such an interpretation is not always necessary. The first practical systems for structural evaluation which are going to evolve will be much simpler. The impedance of a test assembly will be measured before and after exposure to an environmental program and a significant change interpreted as internal failure. One step further will be to establish a "normal" impedance function for a certain structure through measurement on many samples, and then evaluate subsequent production items through comparison with this normal impedance.

Figures 7 through 9 illustrate a case of a small change in a structure which can be detected externally through impedance methods. The structure shown in Figure 7 was tested first when correctly assembled, and then with the circled screw not fully tightened (loose 1/4 turn but still in contact with the lock washer). Figures 8 and 9 show magnitude and phase angle of the measured impedance function, $Z(f)$, in both cases. The change of the frequency of a resonance close to 2.4 kc is apparently caused by the untightened screw.

A more extensive discussion of the use of mechanical impedance for characterization of structures is given in lit. ref. 1 by R. B. Tatge.

Figure 7 Test Structure Mechanically Evaluated Through Impedance Measurement

Detection of Free Objects

Small loose particles such as a piece of wire, a solder ball or a washer deposited by mistake in an assembly during manufacturing, can usually not be found through impedance measurements. The weight of the loose particle might be only a few milligrams, so little that its presence does not change resonance frequencies or damping of the structure. But even this size particle, especially if it is electrically conductive, presents a hazard to the operation of electronic and electromechanical components.

Figure 8 Magnitude of Mechanical Impedance

Figure 9 Phase Angle of Mechanical Impedance

Therefore, a special method has been developed, particularly for detection of small free objects in passive structures. It is inherently less complicated than the impedance method and has therefore come further towards practical applications.

This method utilizes the fact that any loose objects in a structure will bounce around if the structure is vibrated with high enough amplitude. The resulting impacts generate high frequency vibrations which propagate easily through the structure and can be registered by externally located transducers.

The sensitivity of such a system is basically very high. A 5mg piece of lead shot dropped a few inches onto a 1000 lb. steel plate can easily be detected

with an accelerometer located several feet from the impact point. However, practical limits are set by various background signals, and it is necessary therefore to be selective in the evaluation of the impact transients.

Figure 10 shows a typical example of a detection system for loose parts in small electronic assemblies. The test assembly is vibrated with a low fre-

Figure 10 An Electronic Package Mounted for Detection of Internal Loose Parts

quency and slightly above 1g vibration level. The signal is registered by an accelerometer, electronically conditioned, and the desired frequency range then displayed on an oscilloscope screen synchronized with the vibration frequency. Loose parts down to 2 mg weight with any location within the assembly can be detected. From the shape and stability of an impact signature displayed on the scope screen, it can be determined whether it is generated by an entirely loose object or a constrained part such as a flapping wire or cable.

This system, even though basically very simple and not yet developed to the point where the evaluation is automated, has already proven to be very useful. One important application concerns contamination control of electronic packages for space hardware. Several Loose Part Detection Units have been developed in cooperation with General Electric's Missile and Space Division and are today in use for mechanical checkout of sealed electronic packages. There is presently no other method known, which can duplicate their performance.

A Method for Early Detection of Ball Bearing Damage

General

This section will discuss in some detail the development of one particular signature analysis application, a summation system for detection of small defects in operating ball or roller bearings. It illustrates the point that knowledge about failure modes and signature generation mechanisms for the object under study is essential in the development of an effective detection system.

The requirements on this method were that it must detect a malfunction while the bearing is assembled and operating in a realistic environment, and make this detection early enough so that action can be taken before catastrophic failure has occurred. It is easy to realize how much an effective and practical method of this type would increase the maintainability of equipment where rolling element bearings are used.

The first step is to define what failure modes should be considered.

Ball Bearing Failure Modes

In order to select a reasonable model for the malfunction signature generation, it is necessary to know how ball bearings generally fail. Considerable information on this subject is available in the literature (References 2,3). Therefore, no general discussion is presented here. Only some of the fundamental parts are related insofar they affect the choice of model.

Table 1 lists seven of the most frequent reasons for ball or roller bearing failure. If a bearing is properly operated, it normally has a wear rate history

Figure 11 Tested Bearing Type

according to Figure 12. After an initial running-in period the wear rate stays very low for a long time, until fatigue of balls and faces finally results in small scale flaking or cracking somewhere on a contact surface. Once this has occurred the wear rate generally increases and finally results in a failure.

Table 1

Failure Modes for Rolling Element Bearings	
1. Normal Fatigue	Starts as small scale flaking at local spots on the rolling surfaces. Develops into erosion of one or several large areas. This is the normal failure mode for a properly used bearing.
2. External Vibration	Bearings may be damaged both when operating and stationary. Damage usually takes the form of periodic indentation of races.
3. Overload	Generally accelerates the fatigue failure process.
4. Electric Current Passages	Can happen in electric machinery. Sudden high currents cause local craters in balls and races. Alternating currents normally produces regular repeating impressions.
5. Corrosion	Usually starts as local areas with rough surface structure. Develops into roughness and pits distributed over all rolling surfaces.
6. Lack of Lubrication	Increases temperature and wear of most stressed areas.
7. Contamination of Lubricant	Increases wear, may cause corrosion, depending on type of contaminant.

If any of the six conditions except fatigue in Table 1 are present, the long low-wear period is usually shortened, but the wear curve is otherwise similar. The essential fact appears to be that once increased wear has started due to any reason, the bearing is statistically close to the end of its life.

Apparently a malfunction detection system should be designed to react at the time when the wear has just started to increase. The question is then whether any sound producing defect exists at that time. The literature indicates yes, that in most cases the increased wear is physically caused by one or several local defects on either a ball or a race. This defect may be a fatigue crack,

a corroded spot, an electric arc spot, etc. Obviously an impact of some sort can be expected every time this defect spot gets into contact with another surface in the bearing. The problem then becomes to detect this set of impacts while the defect is still small and local and does not yet affect the general bearing performance.

Figure 12 Typical Wear Rate Function for Rolling Element Bearing

Fundamental Malfunctions

We have concluded that bearing failure is generally initiated by local defects in races or balls, the defects caused by any of the conditions in Table 1. This indicates that three fundamental malfunctions should be considered:

- a) Local defect in outer race,
- b) Local defect in inner race,
- c) Local defect in a ball.

With local defect is then understood a single dent, pit, or similar which is so small that bearing performance in terms of drag, alignment, etc. is not affected. (Defect size order of magnitude 10^{-3} inches.)

Any bearing defect can then be considered a superposition of several fundamental malfunctions.

Fundamental Malfunction Signatures

A defect in the inner or outer race of a bearing generates an impact every time a ball rolls over it. Likewise, if the defect is in a ball surface, an impact is generated every time it hits the inner or outer races. If the bearing rotates with constant speed each defect generates a regular set of impacts with an "impact repetition frequency" that depends upon bearing speed and location of the defect. Assuming that no sliding occurs in the bearing, those fundamental frequencies can be computed. The result is:

Outer race malfunction $f_e = \frac{n}{2} f_r \left(1 - \frac{BD}{PD} \cos \beta \right)$ (cps)

Inner race malfunction $f_i = \frac{n}{2} f_r \left(1 + \frac{BD}{PD} \cos \beta \right)$ (cps)

Ball malfunction $f_b = \frac{PD}{BD} f_r \left(1 - \left(\frac{BD}{PD} \right)^2 \cos^2 \beta \right)$ (cps)

where f_r is the relative speed between inner and outer race (rev. per sec.), and BD, PD, β , and n as defined in Figure 13.

OD = Outer Diameter BD = Ball Diameter
ID = Inner Diameter β = Contact Angle
PD = Pitch Diameter n = Number of Balls

Figure 13 Basic Ball Bearing Dimensions

The mechanical signatures which generally can be expected from the three fundamental malfunctions are shown in Figure 14. It is assumed in the figure that the bearing has a stationary outer race and a rotating inner race, and a radial load which is fixed relative to the outer race. Observe that all impacts are not of the same height. The impact size must depend upon the load where the impact is generated and both inner race and the balls rotate relative to the load direction. Each impact is followed by a "ringing," i.e., a structural response which depends more upon the support structure than the bearing itself.

Selection of Detection Method

The proposed model for malfunction signature generation suggests that an effective detection system should look for characteristic transient patterns in the sound or vibration signal which is generated by the bearing.

A special form of summation analysis, therefore, appeared to be the most direct and selective method for detection of regular transient patterns. It recognizes the fact that the background level in most cases is not random

Figure 14 Basic Bearing Malfunction Signatures

noise but rather other patterns, generated by motor, gears, and other noise sources in the machine. A useful detection system must be sensitive to the particular pattern which is generated by bearing malfunctions, but not to other patterns.

A summation system was designed, which performs the following operations:

If $f(t)$ is the raw vibration waveform, starting at $t = 0$, develop the function $F(\tau)$, where

$$F(\tau) = \sum_{n=1}^N |f[\tau + (n-1)T_s]|, \quad 0 \leq \tau \leq T$$

where T_s , T , and N are defined below (Figure 15).

If the summation analysis is done with a certain T_s , it brings out features in the signal that repeat every T_s seconds (or multiple of T_s). Thus, in a bearing application the three impact repetition periods

$$T_e = (f_e)^{-1}, \quad T_i = (f_i)^{-1}, \quad T_b = (f_b)^{-1}$$

are calculated and three T_s are chosen accordingly. The analysis is then performed with those three T_s and possibly some others in between for comparison.

Figure 15

If only the three fundamental bearing malfunctions were present, one at a time in a bearing, and T could always be chosen equal to T_s , the analysis result would look like that shown in Figure 16. In most practical applications, though, some additional repeating features must be expected, like 1/rev. for unbalance vibrations, but they are separable from bearing malfunctions through their different T_s .

Performance Test

To test the developed method in a practical application, a ball bearing was operated as part of a milling machine with a high level of background vibrations. Figure 17 illustrates the test configuration. A vibration signal was registered with an accelerometer located on the support block while the bearing was rotated with 500 RPM under 500 lb. radial load on the arbor. The bearing was tested under two conditions:

- 1) Bearing OK,
- 2) Two dents, approx. $1.5 \cdot 10^{-3}$ in. deep inserted in the outer race (see Figure 18).

The measured vibration signal with the good and bad test bearing was:

- 1) Displayed directly with high time resolution,
- 2) Frequency analyzed with 2% bandwidth,
- 3) Summation analyzed as described above.

The result of the summation analysis for the bearing with the two dents is displayed in Figure 19. Very close to the expected T_s , a significant response can be observed.

*Compare Figure 19

Figure 16 Summation Analysis of Fundamental Bearing Malfunctions
(Assuming that $T = T_s$)

Figure 17 Test Arrangement for Recording of Signatures from an Operating Ball Bearing

Figure 18 Dents Inserted in the Outer Bearing Race. Approximate Depth of Dents $1.5 \cdot 10^{-3}$ in.

Figure 19 Summation Analysis of Vibration Signal from Test Bearing. Local Defect in Outer Race

The three analysis methods are compared in Figure 20 which speaks rather well for itself. Obviously the inserted malfunction is not big enough to show up at all in the direct waveform display or through conventional frequency analysis. The especially developed summation system, which aims directly at a particular type of malfunction signature is the only one of the three methods which is useful.

Conclusions, Future Applications for Mechanical Signature Analysis

The purpose of this paper has been to convey an understanding of Mechanical Signature Analysis, its general scope and in particular its potential for increasing the reliability and maintainability of a vast variety of equipment. It has been necessary to do so by giving a number of examples of various applications, and to tie them together by general comments as they appear needed.

It has been demonstrated that MSA under one headline contains many different techniques, all utilizing sound and vibration analysis to evaluate internal conditions of structures and operating machinery. It is not a question of developing one particular method, but rather to use, for each application, the one technique which is most effective.

However, all the applications which presently show promise are not entirely separated from each other. Together they form a number of groups, where the applications within each group have some common characteristic in the signature generation process. This similarly has resulted in certain general diagnostic techniques which have been found most effective for each group.

With this experience background it is possible in many cases to classify a new problem, and initially select the type of technique which has the best chance of success.

A classification of Mechanical Signature Analysis techniques as it appears today would have the general structure which is shown below.

We can be sure that this classification will change and expand as new applications are recognized and new methods developed, but presently it represents a summary of the major efforts.

Our general conclusion today is that it is possible to develop Mechanical Signature Analysis techniques to successfully perform unique functions in the areas of reliability, maintainability, and quality assurance. The key element in each application has proven to be engineering knowledge about the equipment under study, its normal operational characteristics, its failure mechanisms, and its signal generation and propagation characteristics.

Bearing OK Speed 500 RPM, Radial Load 500 lbs

Local Defect in the Outer Race Same Speed and Load.

Figure 20 Comparison of Three Methods for Detection of a Small Defect in the Outer Race of a Ball Bearing

MSA Technique Classification

		Active Systems		Passive Systems
Group	Cyclic Machinery	Flow Noise Generators	Transient Generators Acyclic Machinery	Structures Assemblies
Most Useful Technique	Summation Techniques	Spectral Techniques	Waveform Display. Generated Summation Techniques. Wideband Spectral Techniques.	Impedance Techniques Generated Impact Evaluation.
Applications	<p>Monitor or Diagnose Motors Bearings Gears</p> <p>Increase Maintainability through evaluation without disassembly</p> <p>Reduce risk for critical failure.</p>	<p>Monitor Operating Characteristics for pumps compressors</p> <p>Detect cavitation stall, early to avoid adverse op. conditions</p> <p>Detect small gas leaks in pneumatic systems or containers.</p>	<p>Monitor or diagnose valves elec. switches for wear alignment sealing</p>	<p>Detect structural failure such as cracks, loose fit through impedance evaluation,</p> <p>Detect loose particles, flapping wires in closed containers such as electronic assemblies, relays, transistors</p>

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Practical Application of the Sonic Analyzer

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INTRODUCTION

Curtiss-Wright began the development of a Sonic Analyzer in 1960. The objective of this program was to produce an analyzer capable of detecting defects in the rotating elements of machinery such as jet engines and transmission systems. Requirements for such an analyzer were that it should be light weight, portable and simple to use. In addition, to insure ease of operation, it should require neither mechanical nor electrical connections to the machine under test. Basically this objective has been achieved.

It would seem that we have arrived at the panacea for all maintenance problems. It should come as no surprise to a group such as this that this is not the case. Let me hasten to add however what we believe is a valid claim for the sonic analyzer; that it is capable of making a significant contribution in the diagnosis of engine condition.

With the present state of the art in electronics, it is relatively easy to build black boxes that will give an indication when certain conditions have been exceeded. Computer capability encourages the amassing of vast amounts of data and allows us to employ sophisticated tests to this data. Computer systems can also tell us if the problem can be fixed at the next scheduled maintenance, if parts are available and what the down time will be. The most important factor in any of these systems is what characteristic of the parameter or group of parameters is used to give an indication that some maintenance action is required.

*Mr. Frarey was an invited speaker but was unable to attend, and his paper was not presented during the conference.

In other words, what is measured, how is it measured, and what limits are established for these measurements.

This paper will discuss the what and how questions for the sonic analyzer and address the question of setting limits.

THE WHAT AND HOW

Almost every rotating element of a jet engine or transmission system produces a discrete frequency signal or set of signals. These frequencies may be calculated from the mechanical characteristics of the rotating element and its rotational speed. For example, a gear mesh frequency is the product of the number of teeth and its rotational speed. In the case of a bearing, three measurable frequencies are generated, (1) a rough spot on the outer race, (2) a rough spot on the inner race and (3) the rolling element.

In many cases these calculated signals may not be pure tones. Fundamental amplitude variations occur and since these signals are not pure sine waves, they contain higher order harmonics. Variations in the relative amplitudes of these harmonics may also be observed. Finally, not all the signals remain constant in amplitude and some display a periodic amplitude variation. A gear mesh signal may vary in amplitude, the period of which is related to the rotational speed of the pinion gear. In these cases, the signal is said to be amplitude modulated.

These variations in amplitude, harmonic content and modulation characteristics have been found to be related to the condition of the part generating the signal. Figure 1 shows an increase in amplitude over a period of time for a signal generated by the rolling elements of a bearing. This signal (left nacelle) is compared to the signal from an identical bearing in another gear box (right nacelle) in which the mechanical condition of the bearing rolling elements were found to be acceptable. For a time period from 63 hours to 86 hours, the signal from the left nacelle shows a measurable increase in amplitude. Figure 2 shows the condition of the bearing rolling elements, after the tear down, at 91 hours. Other examples of component degradation detected by harmonic content and modulation characteristic changes are given in the references.

These variations in signal character are measured by the sonic analyzer and the identification of these critical characteristics has resulted from experimental and theoretical programs conducted during the past seven years. A parallel development of hardware and circuitry has resulted in the design of an analyzer best suited to allow the measurement of these signal characteristics. Operational aspects of the analyzer to be used as a flight line piece of equipment was also considered in this design.

Since the generation of these discrete signals of interest is accompanied by wide band random noise, their separation and measurement requires the resolution of narrow band filtering. Each of the signals of interest may be found at a frequency which is a function of the instantaneous rotational speed. From a practical standpoint, a method must be found to accurately locate the narrow band filter at the proper frequency in the overall spectrum so as to contain the

Fig. 1 Amplitude vs Frequency Spectrograms of the Nacelle Bevel Gear and No. 1 Pinion Bearings at Various Time Intervals.

Fig. 2 Left Nacelle No. 1 Pinion Bearing Showing Spalled Roller.

signal of interest. In addition the location of the filter must be automatically controlled to compensate for variations of rotational speeds, which will occur with time and system rigging.

The sonic analyzer as developed by Curtiss-Wright meets these requirements by utilizing the basic heterodyne principles used in many wave analyzers. Figure 3 shows the block diagram of the analyzer. The local oscillator of the standard wave analyzer is replaced by two components called the phase locked oscillator and the Frequency Ratio Generator. These two components allow for accurate speed synchronization of the "local oscillator" to the speed of the machine. It also provides for precisely locating the narrow band filter at the exact frequency of a given component at this instantaneous rotational speed. This speed synchronization (or locking) may be accomplished by use of the tachometer signal or by acoustically locking to the engine speed. This system allows the narrow band filter to be placed within 2-3 Hz accuracy over the range of 25 to 20,000 Hz. The filter used is less than 15 Hz wide, narrower filters would make speed synchronization too critical at high frequencies, wider filters would destroy the analyzer's ability to detect weak signals in high noise background.

Fig. 3 CWEA-3 Sonic Engine Analyzer Block Diagram.

The analyzer developed as a result of this program is shown in Figure 4. The smaller container includes microphone electronics and provides the option of powering the system from its own internal power or externally supplied 115 V 50-400 CPS power. The larger container includes the basic analyzer circuitry. Microphones are provided as the transducer because of (1) their frequency response, (2) omnidirectional characteristics (which reduces vibration polarization errors), (3) the number of transducers required, and (4) ease of installation. The analyzer design permits its use with other transducers such as pressure sensors and accelerometers.

Fig. 4 CWEA-3 Sonic Engine Analyzer System - Cables and Microphones Connected for Operation.

LIMITS

The limit one sets on any given parameter, beyond which maintenance action is required, is of prime concern in any diagnostic system. This is as true for the time exceeding a given turbine discharge temperature as it is for the sonic engine component signal. Where there is a large amount of reliable data available, limits may be set with reasonable accuracy. In cases where these data are not available, highly controlled scientific tests may be performed on a limited sample. Exhaustive testing of all failure modes and possible inter-

action of components is expensive and time consuming. Less exhaustive testing may produce limits valid only under selected conditions.

A more useful philosophy of limits has evolved during the development of the sonic analyzer. Many faulty components have been observed and related to the sonic signal characteristics. These characteristics have been generally categorized and may be summarized as follows:

1. Bearing signals should be low when compared to the background noise.
2. Gear mesh and compressor blade passage signals should exhibit only minor amplitude modulation.
3. The harmonic content of signals generated by components such as pumps and gear trains should not drastically change.

While this list is not complete, it serves to illustrate the criteria used in establishing the initial limits for the sonic analyzer.

To further illustrate this procedure, the initial application of the sonic analyzer to an engine is outlined here. Mechanical data is obtained on all rotating components of the engine and its accessories. Expected frequencies are calculated for each component as a function of engine RPM. Recordings are made of a limited number of engines to establish microphone locations, to observe signal characteristics, and to determine suitable signals for designing the analyzer plug-in module and for speed synchronization. During the plug-in module fabrication, limits are established for each significant component based on this preliminary data and signal characteristic categories established. The analyzer is then delivered to the field where it is supported by a Curtiss-Wright Sonics Engineer for further development and training.

During this development, if an initial limit is exceeded, the condition of a selected component may be interpreted in the following manner; something is abnormal about the engine with the most likely suspect being that component. This analysis serves three major purposes:

1. It establishes the suspect area of the engine as an aid in troubleshooting.
2. It establishes the depth of maintenance action required. This precludes engine removal when accessory equipment, easily removed, may be the source. (Conversely it prevents the trial and error removal and rejection of expensive accessories when an engine tear down may be required).
3. Once the engine or accessory is removed, knowledge of the suspected area allows practical inspection concentration in that area to provide correlation data.

Operation of the sonic analyzer in this manner partially offsets the costs of establishing valid limits by improved techniques in troubleshooting. In addition,

it allows the user to more directly control limit selection to provide the optimum trade-off between the two uneconomical extremes of too early or too late removal.

Figures 5 and 6 show two engine main bearings detected by the sonic analyzer. During a check out, normal test cell instrumentation did not indicate either discrepancy. The correlation of the condition of these components with the sonic analyzer indications will provide the user and engine manufacturer with a valid basis for establishing maintenance action limits.

Fig. 5 Center Main Bearing Outer Race.

CONCLUSIONS

The use of the sonic analyzer in connection with techniques such as modulation and pulse-shape analysis has proven to be a useful tool in diagnosing mechanical conditions of turbine engines. A significant problem in the development of any diagnostic tool is the correlation of the diagnostic parameters with actual component condition. The best place to obtain this type of data is in actual use. The device may start as only a qualitative assistance to troubleshooting operations and proceed to a confidence level indicator as data is accumulated and rejection limits are refined.

This places a requirement on the diagnostic equipment that it be relatively easy to operate during the early data correlation phases. It should have the flexibility

Fig. 6 Center Main Bearing Outer Race.

to provide initial go-no go limits to aid in identifying engine problems and also provide quantitative data to allow modifying these limits to a particular application. The sonic analyzer meets these requirements.

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Automatic Aspects of Failure Detection Instrumentation

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Summary

This presentation describes a novel data processing technique developed at the Electronics Laboratory of the General Electric Company which shows promise of providing equipment failure detection automatically; i.e., with minimum involvement of human judgment in the end product.

This data processing technique has been called the Frequency of Binary Word (FOBW) technique. In this approach analog signals are converted to a sequence of ones and zeros via an appropriate encoding algorithm. The input analog data is sampled at an appropriate fixed or variable rate. In the case where the "sign" encoding algorithm is used, a one is generated when the polarity of the sampled value is positive and a zero is generated when the polarity of the sampled value is zero or negative. In the "slope" encoding algorithm a one is generated when the present value of the sampled input is greater than the immediate past sampled value and a zero is generated when it is equal to or less than the immediate past sampled value. More complex encoding algorithms and parallel encoding channels can and have been employed. The sequence of ones and zeros passes once through a shift register where by use of "and" gates and counters, the frequency of occurrence of binary words (FOBW) are measured. Binary words are shorter sequences of ones and zeros with "don't care" bits interspersed. This nonlinear process transforms an input analog signal to a point or trajectory in binary word space. The location of this point or trajectory relative to a decision surface in this space is used to provide a decision automatically.

Processing efficiency arises from use of binary words that are sensitive only to signal changes caused by specific component conditions. Implementation efficiency arises from the quick conversion to digital form and the use of integrated circuitry to perform the required functions.

In the design of such instrumentation, a general purpose computer search program is used to evaluate alternative encoding algorithms, the number of parallel encoding channels and the binary words that are sensitive to differences between representative sets of data. This program is also used to assess the performance of various decision rules. In this approach, a system can be completely designed on the computer and the design performance evaluated before any equipment need be constructed.

Because representative sets of data are extremely expensive to obtain in mechanical systems such as a jet engine, an equipment approach was developed for the collection of data during normal operation of jet engines. This equipment has been called the FOBW Data Tabulator. In this equipment the sensor signals are preprocessed to enhance the signal to noise ratio in the spectral region of interest. This is performed by synchronous detection at specific frequencies of interest and low pass filtering. This analog data is converted to specific binary word counts per unit time by a digital data processor. These binary word counts are stored in paper punched tape form and are also printed out in numeric form. The printer output can be used for on-line analysis while the paper tape output is used in conjunction with a general purpose computer "bookkeeping" program to assess the statistical correlation between the binary word counts and actual engine component condition as determined in overhaul inspection.

In operation at an Airlines overhaul site, this equipment monitors specifically installed sensor signals during test cell operation. For a period of time the equipment operates as a data compressor and collector. After a representative set of data has been collected and evaluated, the printer outputs can be used to make decisions.

For a detailed description of this equipment and its performance refer to "A Jet Engine Malfunction Diagnostic System Using Sonic/Vibration Analysis" by A.J. Kasak and L.R. Last, Society of Automotive Engineers, Aeronautic and Space Engineering and Manufacturing Meeting, Los Angeles, California, October 2-6, 1967, paper number 670871.

The FOBW technique has been successfully applied to construction of a ten digit portable word recognizer, a jet engine malfunction demonstrator and to automatic classification of highly identical electromagnetic radiation waveforms. The technique has proven to be highly sensitive to waveform differences. Instrumentation based on this technique results in compact, portable and reliable equipment. Large scale computer programs have been developed to automatically search through data to uncover binary word discriminants for design. Versatile equipment has been developed to experimentally search for and evaluate discriminants of interest for both recorded analog data and on-line operation.

Because of results obtained during the past four years of its application, heavy emphasis is being placed in the General Electric Company on the FOBW technique as a practical means for automatic mechanical failure diagnosis and incipient failure prediction.

Discussion on Papers by Sternlicht, Weichbrodt, and Sollecito

BULLOCK (ORNL): Mr. Sollecito, can you tell me the relative computation times for your general-purpose algorithms as compared to your Frequency of Binary Word (FOBW) technique?

SOLLECITO: I take it that you are talking about implementing the FOBW computation via special-purpose equipment versus the implementation via a general-purpose machine? If a general-purpose machine is available, then you should use it. There are cases, though, where one has a gigantic computer check-out system (for example, for the Apollo vehicle), and there are so many data going into the computer that the data must be predigested. This is really where we started our work; our FOBW technique is a preprocessor to do the initial digestion of the data.

LEGLER (AEC): Are these computer programs generally available from you people, or are they proprietary?

SOLLECITO: The very first set of programs was developed on company funding, and they are proprietary. There is also a second generation of programs developed on a Navy contract, and these are available. We are carrying out some advanced statistical analysis techniques in conjunction with this work, and this information is proprietary; but there is an in-between phase that is available because it was done under government funding.

KRYTER (ORNL): It seems that your FOBW correlation is a modification of a form of digital data reduction which is known in the noise-analysis field as "hard clipping," or "polarity correlation." Are you really doing something different from polarity correlation when you look at either the signal polarity or its first- or second-order derivatives with your binary words?

SOLLECITO: You are right. The polarity-coding algorithm makes use of zero crossings. But the difference is that we look at these binary words to give us the differences between classes of data. It is simple hard-limiting of either the wave form itself, or the derivative, or the integral. We did some correlation analysis based on this principle.

Because the process is so nonlinear, we have experienced difficulty in trying to get a general analysis. We started with a given input, consisting of known amounts of signal and noise, and analyzed the frequency of occurrence of various binary words. Unfortunately we found that the zero-crossing statistics are not preserved through this nonlinear process.

KRYTER: A second question: I think that it is fairly apparent that by looking at only first or second derivatives, or zero crossings, you are throwing away at least a part of

the signal information; what is your justification for doing so? Is it speed in doing succeeding computations as compared to more standard Nyquist sampling techniques?

SOLLECITO: The justification is in the results obtained. One can classify data that have not been seen before. If one takes the standard pattern recognition approach and has a representative set of data, one separates the data into two classes: design data and test data. Then one searches for discriminancy in the design data and finds a classification algorithm that allows separation of those data. Another justification is that a good mathematical model can often guide one toward binary words that are most useful.

DEBOISBLANC (IDAHO NUCLEAR): Mr. Weichbrodt, I have a question on your time-superposition technique. Let us consider the failure of a ball bearing, caused by a foreign particle. If the particle stays in place and has exactly the same excitation as the balls, then it creates a repetitive signal that can be overlayed one on top of another and signal averaged to achieve a gain in signal-to-noise ratio. But, what happens if the particle becomes deformed so as to change the resulting excitation or if it moves around in space and hence is displaced in time?

WEICHBRODT: In this case one would not be able to use time superposition as a means to detect the foreign particle.

We have observed the situation that you described: a fatigue-spall particle comes out of the bearing race or out of a ball and then bounces around. This creates two effects: (1) the hole left by the particle and (2) the particle bouncing around. In this case we have been able to see only the stationary hole where the particle had been before it left the race because it gives a regular signal.

Case Histories of Reactor Incidents

F. T. Binford, Chairman

Case History of the Enrico Fermi Reactor Incident: Anomalous Reactivity and Fission-Product Analyses

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INTRODUCTION

On October 5, 1966 certain anomalous events occurred during the operation of the Enrico Fermi Reactor. These events produced the following signals: the regulating rods were observed to be in anomalous positions, the fission product detector suddenly gave very high readings, radio-activity monitors in the containment building caused alarms to be sounded, and certain core coolant outlet thermocouples were observed to have readings which were higher than should be anticipated for the power level at which the reactor was operating. It has been found that these anomalous events were caused by the melting and displacement of fuel in several subassemblies. It is suspected that the melting was caused by the flow blockage to these subassemblies. Furthermore, it is suspected that the flow blockage was caused by the action of a foreign object which has been found in the lower coolant plenum of the reactor vessel.

This paper is concerned with the interpretation of the anomalous events of October 5, in an attempt to establish the time history and the extent of the melting. The anomalous regulating rod history can be interpreted to provide information on anomalous reactivity effects. The fission product detector information can be interpreted to provide information on the time at which there was a substantial release of fission products. The radio-activity monitor in the containment building provides health physics information on the area and it derived its signal on October 5 from cover gas that had leaked into the containment building. The fission product activity in the cover gas has been analyzed to determine the extent of the melting. Adjunct to this is the fission product activity in the primary coolant, which also has been analyzed for supporting information. The available data on temperature behavior on October 5 has not been included in this analysis since it has been found (Ref. 1,2) that the meaning of thermocouple readings in terms of temperature performance within a subassembly is not well understood due to the possible existence of unanticipated flow effects both within and external to a subassembly which is being monitored by a thermocouple. The proper interpretation of this data must be held in abeyance until certain planned experimental work is completed.

ANOMALOUS REACTIVITY EFFECTS

Analysis

There are three prime data records for October 5 which can be used in the analysis of the reactivity effects. These records are the power trace, from a Keithley micro-microammeter recording a signal from an out-of-core neutron detector, the chart giving the positions of the shim and regulating rods as a function of time, and the charts which give the sodium inlet temperature on the secondary side of the intermediate heat exchangers. The latter provides an indication of the effects of changes in the steam and feedwater system on the primary coolant inlet temperature.

The reactivity which the reactor senses is the net reactivity and the reactor power and the net reactivity are related by the neutron kinetics equations. The net reactivity can be considered to be composed of four components: (1) the feedback reactivity, (2) the reactivity due to the position of the shim and regulating rods, (3) the reactivity due to changes in the coolant inlet temperature assignable to changes in the feedwater system, and (4) the anomalous reactivity. The data available has been analyzed to obtain estimates of the net reactivity and of the first three components. The differences are then ascribed to the fourth component, anomalous reactivity.

The following procedure was used to obtain the estimate of the net reactivity. The Keithley chart was digitized at intervals from 3 to 30 seconds for the period from 1343 to 1520 on October 5. The digitizing interval was selected on the basis of the amount of changes occurring in the power. Normally the interval was 30 seconds; however at one point when there was a rapid power change the digitizing interval was 3 seconds. The data was integrated with the neutron kinetics equations to obtain the net reactivity as a function of time. The basic expression used was that

$$k(t) = 1 - \frac{\phi(0)}{\phi(t)} \sum_{i=1}^6 a_i e^{-\lambda_i t}$$
$$- \frac{1}{\phi(t)} \sum_{i=1}^6 a_i \lambda_i \int_0^t \phi(t') e^{-\lambda_i(t-t')} dt'$$

It was found that a numerical integration for the integral, using a linear interpolation between time points would require a large amount of computer time, since extremely small time intervals would be required to achieve convergence. Consequently, it was decided to perform the integration by piecewise integration with the power being assigned exponential forms between successive time points. Because of the number of evaluations needed in order to specify the time variation net reactivity, the evaluations were performed on a computer.

In a similar fashion the feedback reactivity was obtained. The power data was processed with the feedback transfer function. Since the feedback transfer function is

$$X(s) = \frac{\delta k(s)}{\delta P(s)}$$

therefore one can write

$$\delta k(s) = \delta P(s) \cdot X(s)$$

Through the use of the convolution theorem, this expression can be written in the time domain as

$$\delta k(t) = \int_0^t \delta P(t-t') \cdot X(t') dt'$$

where $\delta P(t) = P(t) - P(0)$. As above, the power was assumed to follow exponential forms between successive time points and the integration was performed analytically. The evaluation was done on a computer. For this analysis the feedback transfer function was assumed to be of the form

$$X(s) = -\frac{0.181}{1 + 0.35s} - \frac{0.0258}{1 + 0.83s} - \frac{0.0516}{1 + 4.0s} \\ - \frac{0.1345 e^{-30s}}{1 + 60s} \quad \frac{\phi}{Mw}$$

The first three terms represent reactor feedback; the fourth represents the feedback due to changes in the coolant inlet temperature when the feedwater conditions are held constant. The 30 sec term appearing in the numerator represents a 30 sec time lag which occurs in the primary system and the 60 sec term in the denominator represents a 60 sec time constant. The power coefficients represent the present best estimates of the total power coefficient and its breakdown for the flow conditions that existed on October 5. The inverse transform of the feedback transfer function is

$$X(t) = -0.518 e^{-2.86t} - 0.0310 e^{-1.20t} \\ - 0.0129 e^{0.250t} - 0.00224 e^{-0.0167(t-30)}$$

This is the form that was used in the computer solution.

The next term considered was the reactivity due to rod positions. The chart for the regulating and shim rod positions was digitized at 30 sec intervals for the period from 1343 to 1520 on October 5. The positions were converted to reactivity values used in the rod calibration curves.

The final measureable reactivity effect is that due to changes in the coolant inlet temperature which can be ascribed to the changes in the feedwater system. For this purpose the temperature traces for the secondary sodium inlet temperature at the intermediate heat exchanger were used. The traces from the two steam generators were synchronized and a reliable average temperature as a function of time was obtained. The temperature was found to vary only about eight degrees from 1343 to 1520 and the variation between the temperatures from the two heat exchangers at any time did not vary by more than one degree. A temperature coefficient of 0.25 cents/F was assigned (that being the isothermal temperature coefficient) to obtain the reactivity effect due to this mechanism.

The anomalous reactivity, as a function of time, then was obtained at a large number of time points as the difference between the net reactivity and the sum of the three ascribable effects at the selected time points, with a normalization to zero anomalous reactivity at 1343.

Results

The power trace during the incident is shown on Figure 1. It is presented here since it is the prime source of information. It should be noted that it is

Figure 1 - Power History October 5, 1966.

smooth and well-behaved. The only sudden changes occurred at about 1512. This made the power trace easy to digitize accurately. Even the sudden changes at 1512 occurred slowly enough so that the times at which changes in power occurred could be pinpointed to within a few seconds. The time history of anomalous reactivity obtained by the method outlined above is given in Figure 2. Three points are evident from this curve. First is that there were no anomalous reactivity effects between 1350 and 1428. At 1428 a general trend of a linear (with time) anomalous reactivity began and lasted until 1455. At that time the slope changed and the new slope held until 1512, at which time the anomalous reactivity began to hold constant for four minutes. Finally at 1516 the anomalous reactivity began to decrease. It is pertinent to relate the associated power levels. They are the following

<u>Time</u>	<u>Power (Mw)</u>
1350	2.5
1428	5.8
1455	17.8
1512	30.0
1516	28.2
1520	26.2

Figure 2 - Anomalous Reactivity History October 5, 1966.

The power has been correlated with the anomalous reactivity in the time from 1428 and 1455 and the results are shown on Figure 3. From this it can be seen that there is a linear relationship, the power coefficient being $-0.5 \text{ } \phi/\text{Mw}$. Since the power level was too low for coolant boiling and fuel melting it is felt that this power coefficient is due to overheating of one or more subassemblies. It was assumed that this power coefficient was in existence throughout the balance of the incident and the anomalous reactivity for the period between 1455 and 1520 was adjusted by the removal of a component equal to

$$P(t) = -0.5 (P(\text{Mw}) - 5.5) \text{ cents}$$

The results are shown on Figure 4 for the overall picture and on Figure 5 for the period between 1510 and 1514. From Figure 4 it should be noted that there is a general trend of about $-0.8 \text{ } \phi/\text{min}$ from 1455 to 1513:30, giving a reactivity loss of about 14.3 cents. Superimposed on this general trend are three transients, two of which are sketched on the figure. The first is a 4.7 cent loss in about 1.3 min at the rate of $-3.6 \text{ } \phi/\text{min}$, followed by a subsequent gain of the same amount at the rate of $+1.4 \text{ } \phi/\text{min}$. A similar transient with similar rates occurred between 1508 and 1510. However only about 1.3 cents of reactivity was involved in that transient. The third transient is the 1512 transient. There are several interpretations that can be put on this transient. One is that the general trend loss of $-0.8 \text{ } \phi/\text{min}$ was maintained throughout the transient. The transient consisted of a gain of $2.3 \text{ } \phi$ at the rate of $+2.0 \text{ } \phi/\text{min}$, subsequently followed by a loss of $4.4 \text{ } \phi$ at the rate of $-22 \text{ } \phi/\text{min}$ and then by a gain of $2.1 \text{ } \phi$ at the rate of $+3.4 \text{ } \phi/\text{min}$, giving a net reactivity change of zero for the transient.

Since the power was not linear after 1455, it cannot be expected that the adjusted anomalous reactivity is a linear function of power. To determine the functional relation, the adjusted anomalous reactivity was correlated with power

Figure 3 - Anomalous Reactivity — Power Correlation 5.5—18 Mwt.

Figure 4 - Adjusted Anomalous Reactivity History.

Figure 5 - Reactivity Transient 1510 to 1514 Hours.

between 1455 and 1500 and between 1505 and 1510. These periods were selected because they did not contain any transients which could destroy any correlation. It was found that the data could be fitted very well by a parabola

$$\rho \text{ (cents)} = -0.12 (P(\text{Mw}) - 19.2)^2,$$

as shown in Figure 6.

Accuracy

The accuracy with which trends in reactivity can be established and reactivity events pin-pointed can be determined by inspection of the data obtained. In the time interval between 1350 and 1428 no apparent reactivity effects were observed. The data scattered rather randomly about a zero reactivity line with a deviation of about ± 1 cent. This is substantiated by similar calculations made on a run that occurred on September 6, 1966. During that run the power was increased from the kilowatt level to about 26 Mw in about 50 min. It is not believed that any anomalous reactivity effects occurred during this run and it was found that the reactivity effects for which there is a known cause were distributed in a random fashion about a zero line with a deviation of ± 1 cent. Therefore the scattering of data points randomly about a trend, where the scatter is within 1 cent, cannot be interpreted since they could be statistical scatter. However if several consecutive points are removed from a general trend in the same direction the existence of a true deviation from the general trend might be believed.

There are several mechanisms which can cause the observed scatter. Listed in the order of probable importance, they are:

- 1) errors in recording and reading the power level
- 2) errors in recording and reading shim and regulating rod positions

- 3) lack of adequate information on the behavior of the coolant inlet temperature
- 4) errors in the feedback transfer function
- 5) errors in the delayed neutron data

Figure 6 - Anomalous Reactivity — Power Correlation 18—30 Mwt.

Interpretation

In the analysis it was found that there was a general trend of reactivity loss at the rate of -0.5 c/Mw from 1428 to 1455 while the power was increased from 5.5 Mw to 18 Mw. The power range in which this occurred is too low to have this effect due to sodium boiling and hence fuel melting and motion. Therefore it probably can be attributed to overheating. It should be expected that there should be an overheating reactivity effect prior to boiling and melting. The magnitude of the effect is quite large, being twice as large as the nominal power coefficient. Therefore, if all things were linear, the core ΔT 's would have to be three times their normal values for all of the core to explain the observed magnitude. It is more reasonable to expect that only a fraction of the core was overheating. Thus, the observed magnitude could be explained by the overheating of a few subassemblies, especially if this overheating would produce outward distortion of the core. The outward distortion of several subassemblies and all of their outlying neighbors along a row or column by about 0.25 in would produce the amount of distortion necessary to account for the reactivity effects of 11.7 cents, which is that which can be derived from the observed anomalous power coefficient. Although there is no direct knowledge that this effect continued unchanged above 18 Mw, this was assumed to be true. As was explained above, this effect was subtracted from the total anomalous reactivity, above 18 Mw, to obtain an adjusted anomalous reactivity. This is the reactivity loss that might be assigned to fuel motion. The net amount due to this effect, 14.3 cents, is consistent with the amount that can be assigned from the sub-assembly examination (Ref. 3). No detailed anomalous reactivity history can be de-

terminated though from the subassembly examination; neither can this be done by the analysis of an analytical model, since no substantiated model is available with sufficient detail.

The causes of the super-posed transients on the reactivity history is a topic for conjecture and there are several models which could cause the observed behavior. Two of the super-posed transients observed could be explained by mechanical processes that would release a large slug of material suddenly. It might have been swept out of the core, only to strike some object and then be swept back into the core. The third could be explained as being caused by anomalous motion of fuel previously melted and swept from the core. A portion of the accumulated fuel material might have broken away from the mass and be swept back into the core. While there, it might have joined with other material of about the same mass becoming molten and be swept out of the core, only to have about half of that total slug break away and fall back into the core. These are possible mechanisms which could have caused the observed effects. They are only examples and need not be considered unique. Other mechanisms probably could be postulated to produce the same effects.

Several days after the incident the reactivity loss was measured by making a subcritical estimate of the base reactivity. It was found by this method that the reactivity loss with the reactor shutdown was 22 ± 3 cents. This value can be compared with the anomalous reactivity, determined by the above method of anomalous to be 26 cents maximum and 23 cents at the time of shutdown on October 5. This good agreement is indicative that either the scram had no effect on the anomalous reactivity mechanisms, other than terminating their actions, or that there were self-compensating effects, i.e. the relaxation of the distortion of the core was compensated by additional fuel motion due to the thermal shock of the scram.

FISSION PRODUCT ANALYSIS

The times at which significant reactivity events began should be compared with the time at which fission products were first released. This can be determined approximately with the information available on the operation of the fission product detector.

Concurrent with reactor operation on October 5, a test was being run to calibrate and evaluate the operation of the fission product detector. The fission product detector receives primary argon cover gas samples, containing xenon and krypton in case of a fuel rupture, from the No. 1 and No. 3 primary pump tanks, the transfer rotor exit port, and the reactor vessel. The gas is introduced into a precipitation chamber which contains a negatively charged tape that collects the solid, positively charged, daughter isotopes resulting from the decay of xenon and krypton. A segment of the charged tape remains within the chamber for a specified soak time and is then drawn into a scintillation chamber, where the decay of the charged fission products that were collected in the tape is measured.

When the reactor was at 1 Mwt, the fission product detector was reading between 250 and 400 cpm with a slight rising trend. These values may be compared with a value of 190 cpm taken at the same calibration settings on September 29, 1966 at a power level of 0.87 Mw.

As the reactor power was being increased, no appreciable increase in count rate was observed until 1359, at which time it was increased from a 550-cpm reading to a maximum of 790 cpm at 1403. Following this, the count rate decreased and settled out at approximately 650 cpm, corresponding to the return of reactor power to 2 Mwt.

The count rate increased from 680 to 1470 as reactor power was later increased. At a reported reactor power of 14 Mwt and time of 1444, the count rate was 1350 cpm. At 1451, some calibration work was done and at 1502, the next reading taken was off of the 5000 cpm range. The 50,000 cpm scale and the 500,000 cpm scale were selected successively with full scale observations on each. A gain check was made to verify that the instrument was performing properly, the results of which indicated no fault in the channel. The fission product detector first was giving an off-scale reading at a time no later than 1502 and no earlier than 1451. Since the detector has a 13 minute lag time, the indicated fission product release occurred between 1438 and 1449, at which time the reactor was at a power of 10 to 15 Mw. The reactivity loss attributed to overheating began at 1428, far before the fission product release. The reactivity loss attributed to fuel motion began at 1455, which is 6 to 13 minutes after fission product release. Thus there was a significant time lag between the time of fission product release and the time at which there was a reactivity loss due to fuel motion.

Qualitative and quantitative measurements of the fission products contained in the primary sodium coolant and the primary argon cover gas were made after the incident. The radioisotopes studied in the cover gas include xenon-133, xenon-131 and krypton-85. Analyses were performed with a vibrating reed electrometer and, where sufficient concentrations were available, a 1024 channel gamma ray spectrometer. Using the latter instrument, xenon-133 was identified as the major isotope present. Early data obtained using the electrometer indicated very high xenon-133 levels with the implication that a large number of pins had melted. Subsequently, it was determined that these data were in error due to the contamination of the inside surfaces of the 1000-cc electrometer ion chambers during gas sampling. Uncertainty concerning the volume of cover gas in which the fission product gases had been mixed also contributed to early high pin failure estimates.

In an effort to resolve these questions, the cover gas was purged into a 2500-cubic-foot waste gas tank from which two 1000-cc samples of cover gas were taken on October 20. These samples were sent to Argonne National Laboratory for analysis of the concentrations of xenon-133 and krypton-85. Based on this analysis, about 45 curies of xenon and krypton were in the cover gas. Using the krypton-85 data and assuming that 100 per cent of the fission gas was released on melting, it was concluded that 71 (average burnup) pins had melted. The data for xenon-133 yielded 54 pins for the assumption of 100 per cent release of xenon-133 and iodine. Higher numbers would be obtained for lower release percentages.

It was concluded from this that melting of the amount of fuel equivalent to that contained within about one or two complete subassemblies is most likely.

A number of radioisotopes have been detected in the primary sodium, including cesium-137, strontium-90, iodine-131, barium-140, and lanthanum-140. Radiochemical separation, beta counting in a thin-window proportional counting system, and scintillation spectrometry were utilized. A considerable degree of uncertainty exists in the determination of the number of fuel pins involved from the analysis of the fission products in the sodium because many of the assumptions of release fractions and plate-out losses for each radioisotope studied are difficult to justify. For reference purposes, a 50 per cent release fraction is considered. The cesium-137 and iodine-131 analyses would indicate that about 31 pins were involved and the gross beta counting would indicate that about 222 pins were involved. The strontium-89 analysis yields a much higher number, 628 pins. Table I summarizes the interpretation of the extent of meltdown obtained by radioactivity analyses of sodium coolant and argon cover gas. The sodium analyses, in general, substantiate the cover gas analyses, i.e., that the fuel failure was equivalent to the complete meltdown of approximately one to two average core subassemblies. This is consistent

TABLE I
RESULTS OF PRIMARY SODIUM AND ARGON ANALYSES

<u>Radioisotope</u>	<u>Concentration In System, Ci (Curies)</u>	<u>Concentration In Average Pin Ci (Curies)</u>	<u>Number of Fuel Pins Melted Per Assumed Release Fraction</u>		
			<u>10%</u>	<u>50%</u>	<u>100%</u>
Krypton-85	1.43	0.02		142	71
Xenon-133	6.4	0.118		109	54
				240**	170*
Cesium-137	2.75	0.17	162	32.4	16.2
Strontium-89	1570	5	3140	628	314
Iodine-131	5.04	0.33	153	30.6	15.3
Gross Beta	5600	50	1110	222	111

* 100% Release of Xe-133 present at time of melting, Oct. 5, 1966 zero per cent release of I-133, which would lead to additional Xe-133.

** 50% Release of Xe-133 present at time of melting, Oct. 5, 1966 10% release of I-133

with the information that has been obtained from subassembly examination work that has been done (Ref. 3).

CONCLUSIONS

From this study of the Fermi Incident, several conclusions can be drawn which are pertinent to the subject of this conference. The first is that the presence of anomalous reactivity can be sensed early and would be a good indication of incipient failure. The second conclusion is that the presence of fission products in the cover gas provides a signal that failure has occurred. However, the time delay in obtaining the signal with the system installed on the Enrico Fermi Reactor is so great that the signal from such a system should be used only as a corroborative signal and not as a primary signal. This instrument does have the capability of providing information on very small failures. The third conclusion is that ex post facto analyses of sodium and cover gas samples provides an indication of the magnitude of a failure incident. However, great care is necessary in the execution of such an analysis. The precision of such an analysis can be greatly improved when further data is available from controlled laboratory experiments on fission product release and fission product plate-out. In general, it can be concluded that the nuclear data available on the Fermi incident has been found to be useful in a reliable analysis of the incident. Similar data taken during future fast reactor operation should be of use in detecting fuel failures in their incipient stages, if the data are analyzed properly by reactor operators. A suitable dynamic analysis program applicable to fast reactors has been developed at APDA (Ref. 4,5,6). This program would perform the analysis of the anomalous reactivity and fission product detector data on-line by the use of a small digital computer operating on-line. It is anticipated that the information developed as a consequence of the Fermi in-

cident will be useful to the designers of systems which can be used to detect incipient failures in fast reactors appropriately.

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The Effects of Subassembly Bowing in the EBR-II

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The combination of highly enriched fuel, small core volume, and strong flux gradients can, under certain circumstances, lead to operational instability in fast reactor systems. Accordingly, a great deal of effort and attention has been devoted to a basic understanding of those feedback processes which could affect the stability and usefulness of EBR-II which, at the present time, is the Nation's only operating fast reactor.

In an effort to establish the stability margin for EBR-II extensive transfer function measurements, supplemented with static power coefficient measurements were conducted under a variety of power and flow conditions during the ascent to power in the summer of 1964. The results of all measurements conducted during the startup indicated conclusively the unconditional stability of the system for all credible reactivity insertions.

As the EBR-II system increasingly assumed the role of an irradiation facility for prototypical fast reactor fuels, the size and composition of the core changed. Such changes were expected to modify the magnitude and time dependencies of the various feedback components. Accordingly, particular attention was devoted to measurements which would reflect any deleterious changes in reactivity feedback. Since the original oscillator rod, used in the determination of the reactor transfer function, failed in October, 1964 it became necessary to place increased reliance on the results of static power coefficient measurements as a criterion of stability. Supplementary information was obtained from studies of the time-dependent feedback recovery following the steep-ramp withdrawal of reactivity.

Since subassemblies containing experimental fuels were less reactive than the driver subassemblies they replaced, corrections for reactivity losses were implemented by installing additional driver subassemblies along the outer edge of the core. Consequently, as irradiation experiments were added, the core was gradually enlarged from the 73-subassembly configuration at startup to 81 subassemblies

in Run 24 at the end of 1966. During this time, the static power coefficient decreased from 3.75 to $3.23 \times 10^{-5} \Delta k/k/MW$. It was also noted that a tendency existed for the power coefficient to decrease during any given run. In any event, such changes were small and up through the completion of Run 24 were never seriously regarded with concern. In all cases the power coefficient was strongly negative.

Upon the completion of Run 24, two major changes in loading were implemented. One concerned the enlargement of the core outline to 88 subassemblies and the other involved the replacement of the two innermost rows of depleted uranium blanket subassemblies with stainless steel-filled counterparts. Upon startup, in April, 1967, it was discovered that for power levels above 10 MW (full power, 45 MW) the behavior of the power coefficient had undergone a significant change, not only in magnitude but in its dependence on power as well. While the overall average power coefficient decreased from 3.23 (at the start of the previous run) to $1.93 \times 10^{-5} \Delta k/k/MW$ the differential power coefficient in the 10-30 MW region decreased from 2.1 to $0.22 \times 10^{-5} \Delta k/k/MW$. Even more disturbing was the fact that continued operation of the reactor was reflected by additional decreases in both the overall average power coefficient and the differential power coefficient in the 10-30 MW region, with the latter value reaching an indicated low of $0.111 \times 10^{-5} \Delta k/k/MW$ near the end of the run. Since this represented an approximate 20-fold decrease, compared with the previous loading, concern was expressed for the stability of the system since the operational characteristics of the system are determined in large part by the differential power coefficient. Inherent in this concern was the underlying fear that unknown mechanical changes were occurring in the system. If such were true, it was reasoned, the possibility of gradual reactivity losses recovering over a short time interval could not be dismissed as incredible.

The long term trend towards lower power coefficient values is believed to be the result of the gradual increase in core size as more and more irradiation subassemblies were added. A study of actual experimental values for the power coefficient as a function of effective core radius (or volume) supports this conclusion. On the other hand, no real correlation between power coefficient values and time has been noted.

The short-term trend towards lower power coefficient values during a run has been identified with expansion effects in the control rods and their shafts. In EBR-II the control rods are suspended from structural features which are essentially insensitive to changes in coolant outlet temperature. The control rods and their shafts, however, are immersed in coolant which has traversed the core. Thus, power increases are reflected by expansion effects which tend to lower control rods with respect to the core. If control rod worths were linear as a function of elevation such effects would go unnoticed but, because rod worths are strongly non-linear, the magnitude of associated reactivity effects depends on the actual position at which the control rods are banked. It is important to emphasize that the extent of rod displacement is a linear function of power and that the actual displacement between 0 and 45 MW is almost entirely independent of control rod configuration. The effect is such that when the rods are banked at a relatively low position, for example at the beginning of a run, rod displacement causes a relatively large reactivity decrease. At the end of a run, with the rods banked at a high position (in the low worth

position of the control rod calibration curve) shaft growth causes only a minor reactivity decrease. Since the associated reactivity effect, whether it is large or small, is a true portion of the power coefficient the power coefficient at the beginning of a run (low banked position, high differential rod worth) will be stronger than it is at the end of the same run (high banked position, low differential rod worth). Fortunately, it is a simple matter to establish the size of the effect from a knowledge of temperature distributions, rod dimensions, and reactivity calibration curves. When such effects are considered, it was shown that the decrease in power coefficient during Run 25 was solely the consequence of control rod displacement and was not symptomatic of some unknown deleterious change in the reactor. The exact origin of the strong reduction in power coefficient at the Run 24-Run 25 interface has not been established. On the basis of circumstantial evidence, however, it seems likely that the stainless steel reflector is involved. Shortly after the reduction in power coefficient was discovered, it was suggested that the stainless steel reflector, located between the fuel and the heavy uranium blanket, was acting as a "window" which permitted higher incident neutron and gamma fluxes at the inner edge of the blanket. Since little heat is generated in the reflector row 6 subassemblies, filled with fuel, and row 9 subassemblies, filled with depleted uranium, ran hotter than the row 7 and 8 reflector subassemblies. Because of this unusual situation, radial heat flow from row 6 into row 7 and from row 9 into row 8 became possible. In effect, inboard surfaces of row 7 subassemblies ran hotter than outboard. In row 8, on the other hand, the situation was reversed. The effects of the reversed thermal gradients were reflected by two bowing attitudes: one inward at the center in row 7, and the other outward at the center in row 8. As a consequence it was postulated that the combination of the two opposite bowing attitudes had the effect of a spring, which forced fuel subassemblies closer together into a more reactive configuration as the power increased. Such strongly positive effects were, presumably, reflected by the strong reduction in the power coefficient following the installation of the radial stainless steel reflector.

Test Reactor Incidents: Selected Case Histories*

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INTRODUCTION

Thermal test reactors such as the Materials Testing Reactor⁽¹⁾, the Engineering Test Reactor⁽²⁾ and the Advanced Test Reactor⁽³⁾ are among the best understood reactor systems. Methods have been developed for calculating in great detail the properties of these reactors throughout their entire normal operating regime and somewhat beyond. The effects of perturbations of fuel loading, moderator conditions, structural components, control elements and changes in experiments have been extensively studied in the critical facilities ETRC⁽⁴⁾ and ATRC⁽⁵⁾. The power distributions, thermal-hydraulics and kinetics are calculable with considerable confidence.

As a result the technical analysis of day-to-day and long term operating problems can be dealt with with a minimum of loss of operating time.

Because of the nature of the mission of the test reactor, it is necessary to be able to operate to the maximum conditions of power density, fuel burnup heat flux and radiation levels which can be justified with engineering confidence. Added to this is a complex geometry with a frequently changing complement of experimental equipment in or near the reactor core. Keeping track of the separate and cumulative effect of these experiments as well as of changes produced by control rod motion, burnup effects and high cross-section fission products is a continual challenge to the technical supporting staff.

Despite all these problems it has been possible to set up and maintain comfortable margins to system failure. Therefore, unanticipated failures have been rare. It has been necessary to study each system failure exhaustively to permit recognition of incipient failures of a similar nature.

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Technical surveillance activities have had an important role in the safe operation of these reactors. The diagnostic techniques have been developed to permit rapid analysis of anomalous behavior.

In this discussion I will discuss two types of incidents which are likely to occur-- flow blockage and power fluctuations.

It will be shown that even severe and sudden interruption of fuel element cooling by foreign matter can instrumentally be limited in effect to only minor core damage, fuel melting and system contamination. In the second class-- power fluctuations-- the nature and source of even very small disturbances can be analyzed using information derived from the reactor instrumentation.

CASE HISTORIES

On December 12, 1961, during a restart of the Engineering Test Reactor following a shutdown during which refuelling had been carried out, an incident occurred leading to melting of part of several fuel elements. The reactor power was being raised to 90 megawatts (full power-- 175 MW) when a power reduction was initiated by the alternate core power measuring instrument which measures the N-16 activation in the process water. This instrument had produced a run-down of the power which cleared at about 70 MW, indicating no further anomalous activity levels, as judged by the N-16 detector. Since the N-16 power monitor was notorious for its tendency to respond to small pieces of radioactive debris in the process water system, the instrument was not considered a fission product monitor, per se.

The instrument was separated from the power reduction circuitry, which was permitted by the Standard Practices in force in 1961. The power was raised back to 90 MW and a search was made for other evidence that a serious problem existed.

The reactor had been operated at 90 MW for about 45 minutes before the N-16 power run-down to 70 MW. Upon return to 90 MW an additional interval of about 35 minutes elapsed during which time the flow recorder indicated a reduction of flow. This was judged to be real when it reached several times the normal noise level of this flow recorder trace. The reactor was manually scrammed.

By this time five fuel elements had experienced some degree of melting. A total of 18 plates, or about 1.5 per cent of the 1100 plates contained in 50 fuel elements and 16 control rod fuel sections contained in the core, were melted over some portion of their length.

Shutdown examination revealed extensive blockage of the flow channels by foreign material which was soon determined to have come from a "sight-glass" which is used to permit improved viewing of the core. This small floating box had an optically clear lucite bottom. It was this material which had shattered and found its way to the fuel element channels. The sight-glass had apparently drifted under the tophead during the shutdown and was not removed before placing the closure plug in place.

A great deal was learned from this incident, which was the first fuel element melting experienced in ETR, about incipient detection of fuel element failure.

It was apparent in an after-the-fact collation of the responses of the various instruments that sufficient evidence existed within a few minutes,

which with information processing methods now available would have quickly verified the conclusion that serious trouble existed.

The recorded flow rate, properly interpreted, exhibited a systematic loss of about one part in 500 per minute, which, taking into account the root-mean-square noise level which was about 2 per cent, gave positive indication in about 10 minutes when the record was analyzed after-the-fact. However, the operating staff accepted the evidence as definite only after the flow had changed by a factor of about 3 times the noise level. The decision to scram the reactor was based upon this and confirming evidence from the neutron level instrumentation. The total time taken for this analysis was between 30 - 35 minutes. However, it should be remarked that the reactor was held at half power while the various bits of data were being put together. In the vast majority of cases, the fuel elements are operating cold at half power and full flow.

The operating practice which permitted the removal of the N-16 detector from the power reduction circuit was revised to prohibit this step. Although primarily intended for power measurement, and even with an inherent 5 second delay due to water transit time and counting statistics, this instrument is very sensitive to anomalous radioactivity in the process water system. It also has been confirmed that normal good housekeeping keeps the system clean enough that the loss of operating time because of this instrument is negligible.

Figure 1 shows the top view of one of the fuel elements showing the magnitude of the flow blockage. From the flow measurement this must have occurred progressively over many minutes. Only about 20 per cent of the fuel core in this element reached melting temperatures in this incident.

The second occasion of melting in the ETR occurred on February 20, 1967. In this case the blocking agent was a strip of adhesive tape which had been left in the tank during the shutdown. Ironically, this tape had been used to hold film around an inpile tube to permit a radiograph to be obtained of a weld joint-- a safety requirement. Because of the aluminized backing of this tape it went undetected during the checkout before closing the tophead.

The reactor had been operating at full power (175 MW) for about 13-1/2 hours when an automatic power reduction was initiated by the N-16 detector. This event occurred so suddenly that melting was inevitable. However, the extent was very effectively limited to the loss of only 2 per cent of the contents of this element by the N-16 detector. This confirmed the judgement made six years earlier that this instrument is a very good incipient failure detector. Since the superior behavior of the UAl₃ (aluminum powder metal-lurgically prepared fuel elements) during melting allowed the cladding to slump and release fission products while the matrix retained appreciable structural strength, the cleanup of the reactor was accomplished very easily. Although a substantial fission product inventory existed in this fuel, the reactor was returned to operation within two days.

The close examination of various instrument records showed no observable flow disturbance or fluctuation in the neutron level system preceding the power reduction. In the ETR the void coefficient is quite low in the core and hence the amount of void required to indicate above the normal noise level upon neutron channels is about 20 times as great as in the Materials Testing Reactor where incipient boiling can quite easily be observed by monitoring the neutron level detectors.

Fig. 1 Flow Blockage in ETR Fuel Element
Caused by Plastic Sheet

In looking at these two melting incidents from the point of view of this conference, it can be said that more rapid instrumental interpretation of the fission product signal and a full scram will protect this class of reactor against a very severe, instantaneous flow blockage, and will limit fission product release to tolerable levels. The behavior of the uranium-aluminum intermetallic fuel matrix contributes to this conclusion by limiting the fission product release even with severe melting. (Figures 2 and 3)

The third ETR operational "event" did not reach the level of "incident" because the nuclear instrumentation gave evidence of incipient failure.

Fig. 2 Top View of ETR Fuel Element Showing Flow Blockage by Tape

In this case, which occurred in the cycle presently underway, a cyclical neutron level fluctuation was observed upon all neutron levels and neutron period channels with a frequency of about 4 cycles per second.

By varying the reactor flow rate and power level it was soon determined that the effect was neither due to boiling nor vibration of the reactor components. Variation of the flow in the inpile experiments soon disclosed that vibration of the inner flow tube of a pressurized water loop was giving a reactivity disturbance. Considering that the amplitude of the reactivity perturbation was only $0.0002 \Delta k/k$ and that no special instrumentation was required to disclose that an abnormal condition existed, it is clear that the monitoring of the neutron level provides an easy indication of a very small cyclical reactivity perturbation.

By digital processing of this signal, as will be done in the Advanced Test Reactor, it is expected that disturbances of the order of a few parts per million in $k_{\text{effective}}$ will be observable.

Fig. 3 Melted Plate of ETR Fuel Element
During Flow Blockage Incident
(Note Remaining UAl_3 -Al Matrix)

CONCLUSIONS

It can be reasonably concluded that, for thermal test reactors, digital processing of signals from water flow, system pressure and differential pressure, radioactivity monitors, and neutron detector signals should provide a combination which will effectively protect these reactors against incipient failures.

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Discussion on Papers by Klickman, Smith, and DeBoisblanc

FRY (ORNL): Mr. Smith, could you tell me why it took so long to replace your failed rod-oscillator? Would this oscillator have helped you to diagnose your bowing problem?

SMITH: We have not had much luck with oscillator rods in the EBR-II. There is quite a distance separating the core and any region where an oscillator rod drive could be located. We have had an oscillator rod in the reactor since 1964; but that rod began to bow, and we took it out early this year. The present rod seems to be working satisfactorily over a limited frequency range, but it too bows when the reactor power is increased. Apparently the boron carbide column bows and causes the rod itself to bow and seize in the thimble.

In answer to your second question, having the rod operational would have given us more definitive information much earlier and would have helped us in our diagnosis. In my opinion there is no substitute for transfer function measurements to detect fuel-element subassembly bowing.

BULLOCK (ORNL): Mr. Smith, you said that the delayed power coefficients have been changing with time, but you did not say anything about prompt power coefficients, which I presume would be of more concern from a safety standpoint. If there are delayed effects, how fast are they, and do they alter the cold-slugging accident and the startup accident?

SMITH: I restricted my remarks to the static-power-coefficient measurements, and these would include all prompt, intermediate, and delayed components. We have been using a rod-drop method for some years. We drop a special control rod, worth about 5 cents, and record the power trace with a magnetic tape recorder. This information is digitalized and then put through an inverse kinetics code that evaluates reactivity as a function of time. Knowing the worth of the rod and knowing the reactivity in the system, one can subtract the rod worth from the gross reactivity and arrive at the time-dependent feedback recovery.

We have used this method to search for changes in the prompt power coefficient, and we have not been able to detect any change in it whatsoever. It was for this reason alone that we felt justified in continuing to operate the reactor. Had we seen any change in the prompt component, we would have been greatly upset because this is one of those mechanisms in a fast reactor which one relies on to help fight any sudden reactivity insertion.

We have recently refined this rod-drop method, and I predict that with these improvements we will be able to observe very prompt components having time constants of the order of 30 milliseconds.

**Conclusions on the Achievements
of the Conference**

Moderator: C. J. Borkowski (ORNL)

Panelists: H. L. Balderston (Boeing), E. P. Epler (ORNL), F. C. Legler (AEC),
A. Pearson (AECL), D. P. Roux (ORNL), F. Schroeder, Jr. (PPC), and
M. A. Schultz (Milletron)

BORKOWSKI: As most of you know, this conference was organized at the request of the Atomic Energy Commission, and the objective was to determine the state of the art in the area of detecting incipient malfunctions of systems. We have heard in the past 3 days not only papers from the nuclear field but also from the nonnuclear aerospace and other areas.

What we would like to accomplish this afternoon is to review what has transpired in the last 3 days, to get impressions from each of the panelists as to what was achieved at this conference, and, perhaps more important, to decide where we should go from here. What should we do as a follow-up?

Since Mr. Legler formally opened this conference, I would like to ask him to begin the discussion.

LEGLER: First, I think we should give the ORNL people a hand of applause for the job they have done in organizing this conference. I personally believe that we have fulfilled our conference objective, which was to hold up to view some of the new techniques, concepts, and equipment that are becoming available to be used in incipient failure detection diagnosis. If we did not hear about every new technique, we must realize that some companies have proprietary information that they are understandably not willing to release ahead of doing business with facilities.

Why is the AEC interested in incipient failure diagnosis? In the previous session you have heard some good reasons for our concern for high-performance reactors, such as the EBR-II and Fermi. Unavailability means loss in research and development, thus affecting the progress of the entire program. The AEC has been a paying partner in the past in many reactor demonstration plants. Many of these plants have been disappointing because of too much downtime. Our director does not want this to occur in the future.

In 1973 the AEC expects to have on line a very high-performance, liquid-metal, fast-fuel test facility, and I hope that the people from Pacific Northwest Laboratory and all the industrial contractors who are going to be connected with that project can use some of the techniques that have been described at this conference.

In addition, before 1973, the AEC, together with the U. S. reactor and manufacturing business, will probably start construction of a large liquid-metal-cooled fast breeder reactor (LMFBR) plant. I trust that you people from the LMFBR Program Office can factor the findings of this meeting into the nation's LMFBR program plan. This also applies to the manufacturers who are going to be a part of the LMFBR program.

BORKOWSKI: I will now ask Mr. Schroeder, who so eloquently set the theme for this conference, to tell us how well we adhered to his suggested directions.

SCHROEDER: One of the objectives that I noted was that of giving all of us an increased awareness of some of the things that are going on in this field, with particular emphasis placed on picking up information from those outside the reactor field. I think that we will all agree that from this cross-fertilization standpoint our time has been well spent here, and we have accomplished quite a bit in that regard.

I recall making the prediction that the interpretation of data received from sensors which provide indications of incipient malfunction would appear to be a much bigger job than the mere acquisition of appropriate signals and indicators. In my own mind this prediction has been amply borne out by the discussions that we have had on detection and interpretation of the information. In other words, in their present state of development, many of the devices that people are working on merely tell that there is something different about a system today as compared with yesterday. There may or may not be something truly wrong with the system. We are faced with a complex detective job to figure out just what it is that is different and whether it is significant and what we ought to do about it. I am a little discouraged; I think that these techniques are going to have to yield unambiguous answers if they are to help designers meet the criteria of the regulators for inspection and testing. I think that we must have devices that tell us not only that something is wrong or different but also what the specific trouble is and whether it is significant. This is particularly true when we talk about the decision to shut down a large commercial power plant, with accompanying economic sacrifice. The operator needs assurance that he is shutting the plant down for good reason.

This leads up to the next comment, which is that there still is quite a gap between the techniques being developed and investigated in the laboratory and real on-line systems that can be handed over a production basis to a plant operator.

Maybe part of the fault here is with the user. I think that Mr. Sowards pointed out yesterday the need, in terms of engineered safeguard systems, for a systems engineering approach, which includes as part of the design effort an identification of those components or subsystems which must be tested and inspected. One can then make allowances for this in his design, and perhaps a little more effort in that direction would lead to more specific objectives upon which the people developing incipient failure techniques can set their sights. Such definition of objectives might accelerate the bridging of this gap between the laboratory and the field.

I stated on Monday morning that another objective of this conference should be to provide some indication of what additional research and development work is appropriate and in which areas we should be spending more money—or less money. I do not believe that we can adequately answer these questions until we have come to grips in a more definitive way with what the targets are.

BORKOWSKI: Still following the sequence of speakers on Monday, I should like to call upon Mr. Schultz next.

SCHULTZ: First off, I would like to remark that the conference organizers have gathered a most interesting group of cross-fertilized speakers. I have certainly learned a lot in these 3 days.

Now permit me a few criticisms. I wish to expand on some comments by Mr. Pressesky on Tuesday morning, in which he said that there are a lot of very interesting schemes starting to float around and that we as research people have some responsibility to get these techniques and devices incorporated into the power plants. Some of these schemes (our neutron noise detectors, for example) have been with us for a number of years, and the question is: What are we going to do about it? How do we get the word out? In particular, how do we feed back some of this information to the designer? This is an incestuous conference if we just sit here and pass this information among ourselves and do not get it back to the power-plant people—either the operators or the designers.

It seems to me that we can attack this problem in one of three ways. First, we can ignore the problem and ultimately it will be taken care of. Second, we can wait for the application of pressure by the regulatory people. You can be assured that this will ultimately happen. The existence of this conference is direct proof of the fact that the regulatory people will get into the act if we do not move of our own accord. Third, and

what I consider to be the best course of action, is the good old American idea that all of us must get out in the industry and sell these ideas to the proper people. Those present from the design companies must go back and tell their people that these tools are available. And all the rest of you, particularly those in the national laboratories, must realize that you have a responsibility to try to get your developments out into industry. So far we have only partially succeeded in this area, and we have just got to sell more effectively.

This brings up another point. The title of this conference has "Power Plants" in it, yet there is not a single utility man in the house. Surely the plant's ultimate operator has something to do with, or at least has an interest in, incipient failures. In this same context, while there are over 50 power reactors under design, being constructed, or in operation at the moment, I find that only 11 people out of the 100 attendees at this conference represent the companies who design and build reactors and entire power plants.

One final criticism: this conference has largely ignored the human as a part of the system, in the sense that the orientation has been toward instruments. People, of course, can be regarded as a portion of the incipient failure measuring device, but it remains a fact that a lot of system failures are still caused by people. Have we considered that there may be some kind of scheme for measuring incipient failure of people? If a man comes to work in the morning obviously red-eyed, maybe he should not be allowed to sit at the reactor control console. For a future conference cast, it might just make some sense to include a psychologist or a medical doctor. Maybe we can do something about incipient failure in people.

BORKOWSKI: I would now like to ask Mr. Pearson from Chalk River Nuclear Laboratories to give some of his impressions.

PEARSON: First of all I would like to comment on the first session of this conference. About 7 or 8 years ago, a similar conference was held at Gatlinburg in which we discussed things like redundancy, coincidence, and on-line testing. At that time the principles seemed fairly well developed, and those of us from Canada went back home and put the ideas into practice with a great deal of success. One of the reasons that we adopted these redundancy schemes at that time was so that we would not have to look for incipient failures. In those days we were most concerned with catastrophic failure. I make these remarks because my Canadian colleagues and I did not bring to this conference an adequate presentation of our work along these lines, partly because we misread the title and thought that the topic of on-line testing would not be covered at all. However, I see in retrospect that it might have been useful had we presented some case histories, demonstrating how useful on-line testing has been over the past 8 years.

Now, with regard to the rest of the sessions, I think without a doubt that the most interesting session was the one on acoustic emission, and we will take that subject back to our own laboratories to think about. The acoustic emission business looks to me as though it is in about the state that nuclear spectroscopy was 15 years ago, and I think that we have another 5 or 10 years of research work ahead of us before we will have installed instruments.

I myself do not have too much patience with those members of the audience who needle the researchers into being practical. I think that it is quite decent to be impractical in research. And there seems to be a lot more research work to be done, both in the area of acoustic emission and in the measurement of reactor noise, before one can really make a working instrument based on one of these principles.

BORKOWSKI: I believe Mr. Sowards is next.

SOWARDS: It is somewhat difficult to make remarks that are applicable to all sections of the conference because I think that the status is vastly different in the different areas. For example, it appears to me that, in the area of basic reactor instrumentation and reactor protective circuits, we are far ahead of the corresponding state of the art in incipient failure detection in the primary loop and in some of the other parts of the system. If we look at the history of the reactor business, we see that this is quite logical because we started out by worrying about the unique characteristic of a reactor—its very fast

transient power capabilities—and this is where the emphasis has been put. However, a definite shift in emphasis is now necessary to conform to the overall present-day needs.

With respect to basic technology, in terms of the techniques that are available, we have good reason to be optimistic regarding the future because there seem to be enough observable physical phenomena to give us good communication with a large number of the sources of failure. However, in terms of what practical, useful, and reliable application we can make of these techniques today, I do not think that the situation is particularly healthy, and this is the area in which we need to place most of our emphasis.

Where do we go from here, and how do we get there? I think that we need to look at the specific problem at hand, the power-reactor situation, and develop the requirements of the system in terms of its application. What specific faults or incipient failures do we really want to look for? Reactor boiling? The indications of internal core structural failures? The primary-loop incipient failure? Foreign bodies in the primary loop? Then, on the basis of such decision, we can develop some system requirements to feed back into the research and development programs.

I do not completely concur with Mr. Pearson's viewpoint that the research and development man can, of intention, be impractical. If these research and development efforts are to produce useful results, they are going to have to have some direction in terms of specific goals; and I think that, if this were provided, we could then put together the best capability that today's technology can give us for looking at whatever spectrum of incipient failures we want.

I also think that it is important at this stage of the game not to make any rigid rules regarding interpretation of results or courses of action demanded by particular results. Rather, I think that we should just regard the results of these advanced tests as additional pieces of information because, in effect, all these devices are really extensions of our own senses. They give us a better set of ears, in effect. With their help we are able to probe into the reactor's structural behavior, its primary loop, and determine in a more sensitive way what is going on inside the machine. This approach will then tie the research and development effort more closely to the useful application.

BORKOWSKI: Mr. Roux, do you have any impressions to offer, either as an attendee or from the Conference Program Chairman's point of view?

ROUX: In my opinion we have succeeded, if one considers this conference as a starting point, in exchanging information on the overall subject of incipient failure detection and diagnosis. However, we still have a long way to go. We have to do much more research and development work to implement these techniques before they can become applicable to power reactors. Also, we have to sell our research and convince the reactor industry that our incipient failure diagnosis techniques are good, and I strongly agree with Mr. Schultz on this point. Now we are faced with the fact that exotic on-line testing equipment is expensive. The reactor industry will certainly think twice on the value of incipient failure detection of components and systems before investing a large amount of money.

As Conference Program Chairman, I would like to comment on two points. First, should we hold another similar meeting related to the same techniques? My opinion is "yes," for in approximately 2 years enough progress will have been made to reconvene. Second, I have been disappointed by the relatively poor response and participation from the private reactor industry and the power utilities to this conference. We will have to lay a better groundwork next time to attract them. Hopefully the proceedings of this conference will convince them to look closer at the matter.

Finally, I would like to thank our friends who came from outside the reactor business. Their attendance was very valuable to us. I hope that they will feel the same way, having learned about our problems and our techniques.

BORKOWSKI: Mr. Epler?

EPLER: I have observed for many years that the reactor instrumentation seems to lead the field in certain areas, perhaps because people look at instruments. That is what they are for. And the protection instruments have led the field for many years in techniques for assessing reliability and performance. The on-line testing is pretty highly developed. Also, as I have said many times, the attention to these matters diminishes as some exponential function of the distance you are from the center of the core.

In this conference we talked first about on-line testing of protection instruments and then we talked about the primary system. We did not get very far in the primary system, but, similarly, people did not really get interested in containment closure until somebody said, "Well, look; you have got to make it work." Now I assert that this situation still exists, and I think that a part of the output of this conference must be to sell these ideas so that they will be incorporated into designs before we are forced to by higher authority.

At ORNL we have been doing incipient failure detection of a somewhat unusual sort for many years, and very successfully. In the ORR we are still using some antiquated vacuum-tube circuitry in a protection system having very fast response (of the order of 10 or 12 milliseconds), and we are not using the usual coincidence; i.e., any one of several channels can scram the reactor, including the much-maligned period scram. The reason that we can tolerate this delicate equipment and this lack of coincidence, and yet operate with much less than 1 scram per year as a consequence of instrument malfunction, is because we have instrument maintenance people who can "eyeball" incipient failure. It is somewhat miraculous to me, but they understand something that is noisy in the signal. They see a trace on the chart that looks just a little bit different from normal, and so they get in there and fix the instrument before it progresses into a full failure.

Now this is easy to do in a dynamic system, and it is not too hard to do in a static protection system. So we incorporate on-line testing into our protection system to make sure that something does not progress to a complete failure and stay failed. In the HFIR system, for example, we have now racked up over 30,000 successful system tests, in which we tested the system from end to end. During this time we have not had one clear-cut case of an unsafe failure. We have had small troubles in the instruments, but somebody always noticed the trouble and fixed it before it could develop into a situation that would be discovered only when testing the system.

So I would say that we are a little bit ahead of schedule on reliability of reactor protection systems, and I think maybe we might be doing ourselves a disservice if we do not shift our attention to the outlying parts of the plant.

Now in the area of engineered safeguards, I think that it is quite apparent that we are behind the times in utilizing well-known techniques that have been developed through the years for the reactor protection system. Furthermore, the engineered safeguards have a peculiar problem compared to the protection system, which has to work just once. In contrast, the engineered safeguard may have to work continuously for months in an environment that a human cannot approach for maintenance. To make matters worse, one may not be able to predict this environment when one designs the system.

So I challenge the industry to use the techniques that we have seen today. I am impressed with the light-pipes and with the noises and squeaks that people can interpret. Let us see if we can find some way to incorporate some of these techniques in the engineered safeguards which have not yet been called on to work following an accident and which may be called on to run for months, perhaps, without repair.

BORKOWSKI: Now let me hear from the nonnuclear member of the panel, Mr. Balderston.

BALDERSTON: My observations of the conference indicate that there is a great deal of commonality between the problems of the aerospace industry and those of the nuclear industry. The nuclear industry cannot possibly risk catastrophic failure, and its capital investment is extremely high. Nevertheless it has to compete costwise with coal-burning plants; it has to keep the kilowatt-hour cost down to where the public can afford to pay for it. So there seems to be some sort of limitation on the costs of these systems that we are going to try to install in an effort to detect incipient failure. However, I think you will find, as you get into this program a little further, that perhaps you can find simpler incipient failure detection systems than you presently anticipate.

In talking about where to go from here, I have not heard much about peripheral equipment: the generators, pumps, turbines, and switchyard. Certainly malfunctions in these units must have some effect upon the reactor which can be dangerous or derogatory in nature, and I think that we should attempt to look for incipient failure of this peripheral equipment along with the nuclear reactor itself. I think that one should consider the

plant clear out to and including the switchyard. I think that the peripheral equipment can be instrumented for incipient failure detection in short order because the techniques for those types of equipment are pretty well established and very well documented. The devices for doing these things exist on the shelf. There are established techniques (walk-around techniques and automated techniques) which can be used on pumps; on rotating, reciprocating, and maintenance equipment; on structural components; on pipes; etc.

As for the reactor itself, in the meantime you could study it more thoroughly and make a selection from the many excellent techniques that have been discussed. I realize, of course, that applications research will be essential to utilize many of these techniques in your particular case.

Also, I think that as one goes down the road in the application of these techniques, some consideration must be given to the data-analysis system as a commonality. Because if one gets involved with all the different techniques that we have been discussing here and with all the different processing procedures related to them, it is possible that one will have built a number of different processing systems; whereas if one would use the technique just a little bit differently, one could obtain the same results more efficiently by utilizing a common data-processing system.

I also recommend that we should get together again when the progress that we have made makes it worthwhile. Two years from now sounds reasonable.

I certainly believe that the information and opinion exchange has been beneficial at this meeting and will probably be more so at the next. Also, there is help for your problems in the aerospace industry and in the industries that are currently providing the aerospace industry with technical support in sonics, supersonics, and the many other techniques that have been discussed. There are many people who can give help, who can actually show how to instrument your plant.

BORKOWSKI: Do any of the other panel members wish to comment on what they have heard so far, or shall we turn the discussion over to the floor? Since there are no immediate questions, I would like to ask Mr. O'Kelly from the AEC if he would care to comment on whether we have met some of his objectives and interests in this conference.

O'KELLY: I am a member of the Advisory Committee on Reactor Safeguards (ACRS), and I have been associated with research all of my professional career. I think that I have the capability of understanding good research work and of recognizing it when I see it. I am happy to say that I have detected its presence during the last 3 days, and I am very glad I came to this conference in representation of the ACRS.

Now there is a great danger in picking out some specific topic in a meeting like this and saying that it is of particular value, and so I will not try to do that. I think that there are a great many things which have been discussed here that could be of real value. I must agree with Mr. Schultz, however, that it is no good if you cannot sell it! My professional career has told me that, and I think that this is one of the criticisms leveled at research by industrial management all the time. While I cannot speak officially for the entire ACRS, it is my personal opinion that they would be unhappy if something concrete did not come out of this meeting or others like it to follow.

The thought of waiting 2 years for a follow-up conference is rather disturbing to me because it looks like the practical effects or results are not too far away in some of the techniques described. I will close by saying that the ACRS will be watching your progress, and they will be helping the Division of Reactor Licensing to get another meeting going one of these days. We wish further success to all of the fellows who have reported their work here.

BORKOWSKI: I wonder if Mr. Boyd, who is a member of the Atomic Energy Control Board of Canada, would like to make some observations?

BOYD: The Atomic Energy Control Board is the regulatory body in Canada, and so I represent the Canadian equivalent of the Reactor Licensing Division in the United States.

I personally have enjoyed the meeting and have found it very interesting. The work, I hope, will be fruitful.

But I would just like to remind you that, from the regulatory point of view, what we are really interested in is "assurance of quality." This is a broader area than incipient

failure detection, and it stems right from the very beginning of design concepts right through to operations. As a quick example, if the requirements for the design are not properly thought out, the plant will probably fail, regardless of whether you have manufactured it properly and whether you have tested it properly.

I would also like to reemphasize that the operator, as has been alluded to previously, is a component in this total question. In certain instances, operators have been causes of accidents, but on other occasions (mentioned by Epler and DeBoisblanc, I believe), operators have been good detectors of incipient failures, and any of us who have been involved with the operation of reactors can point to many instances of this. In fact, I have the feeling that some of the devices that were described here are more sophisticated ways of doing what a good operator has been doing by himself in the past.

In conclusion (and I remind you that I am speaking strictly from the dogmatic position of a regulatory person), it is my opinion that the application of any of the techniques that have been described at this conference would not lead to any changes in regulatory requirements. Their application could lead to additional assurances, yes; but, in their present stage of development, I doubt very much whether they could lead to any relaxations in other areas of regulatory restrictions.

BORKOWSKI: Thank you gentlemen from the panel, Mr. O'Kelly, and Mr. Boyd. I would now like to ask if there are any criticisms, observations, or impressions that anyone from the floor would care to make?

BULLOCK (ORNL): From the standpoint of motivating industry to install and use some of these failure diagnostic techniques, I would like to ask a question of Mr. Klickman or Mr. Nims. What do you think would have been your management's reaction to a proposal for a reactivity balance system such as you are now installing if you had made the proposal prior to the meltdown incident at Fermi last October?

NIMS (APDA): I think it is very likely that, if we had put together a decent analysis and indicated the value of it, with the proper approach our management would have been convinced it was worthwhile doing. I think that the whole thing has to be eventually put in terms of dollars and cents because the utility companies can appreciate a proposal on that basis. In connection with the light-water reactors, for example, they are installing on-line computers where they can see that an economic incentive exists.

Along the lines of constructive criticism and suggestion, I think that it would be a good idea at the next conference to include electric utility companies: people from Saxton, Dresden, Yankee, Big Rock Point, etc. Also, I think it would help tremendously if the whole conference would be started off with a session on "Episodes and Situations in Power Plants," in order to put some of the more detailed topics into better focus, to see where they fit into the actual problems that occur, and to give a little better feeling for the benefits to be derived.

BALLARD (SANDIA): The word "educational" has been mentioned several times during this conference. Personally, I think that in the weapons business our problem has been to educate the subcontractors on the use of known, valid methods of nondestructive testing at the component level and at the subsystem level. We have found the same kinds of human errors that you fellows mentioned today. All the incidents, episodes, and situations that were quoted in this afternoon's session involved a human error that perhaps only education can help. On the other hand, Mr. Epler mentioned the fact that a qualified operator probably has been our real safeguard on many occasions, rather than the instrumentation that we have provided. And I say that the same thing is certainly true in the field of nondestructive testing. A qualified radiographer or an ultrasonics man (one that is educated both to the strengths and limitations of the method) can be the best tool, the best person in the whole procurement chain.

One final remark: If there is a follow-up meeting, I would suggest that more attention be paid to the subject of component quality. We need to get down to the things that fail, the items on which we should certainly be able to put good specifications and good standards. At this conference, most of the talk has been oriented toward complete systems. Well, systems do not fail unless some of their constituent components fail, and so there should be more talk about component failures, component standards, and the application of today's technology to these subjects.

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