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Twelfth Water Reactor Safety Research Information Meeting

Volume 6
- Plenary Session - II
- Human Factors and Safeguards Research
- Health Effects and Radiation Protection
- Risk Analysis
- EPRI Safety Research

Held at
National Bureau of Standards
Gaithersburg, Maryland
October 22-26, 1984

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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Compiled by: Stanley A. Szawlewicz, Consultant

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ABSTRACT

The papers published in this six volume report were presented at the Twelfth Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland during the week of October 22-26, 1984. The papers describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included twenty-six different papers presented by researchers from seven European countries, Japan, and Canada.

PROCEEDINGS OF THE
TWELFTH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING

October 22-26, 1984

Published in Six Volumes

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held at the

NATIONAL BUREAU OF STANDARDS
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PREFACE

This report, published in six volumes, contains 176 papers out of the 205 that were presented at the Twelfth Water Reactor Safety Research Information Meeting. The papers are printed in the order of their presentation in each session. The titles of the papers and the names of the authors have been updated and may differ from those which appear in the final agenda for the meeting. The papers listed under the session on Human Factors and Safeguards Research did not appear in the agenda but were prepared for the panel discussions that made up that session.

REMARKS BY

COMMISSIONER FREDERICK M. BERNTHAL

U.S. NUCLEAR REGULATORY COMMISSION

TO

TWELFTH WATER REACTOR SAFETY RESEARCH

INFORMATION MEETING

ON

OCTOBER 26, 1984

IN

GAITHERSBURG, MARYLAND

I'm honored to join this distinguished assembly for the Twelfth Annual Conference on Light Water Reactor Safety Research.

I stand here keenly aware that, by actual count, this is the 198th speech you've been called upon to sit through this week, and there are 28 more speeches still to come. So I will repair to the good advice on speech-making which President Franklin D. Roosevelt once gave his son: "Be sincere, be brief, be seated."

I can say with the utmost sincerity that I admire the good work which scientists and engineers from the many nations represented here have done -- and continue to do -- in the important field of safety research.

The days are long gone when virtually all of this kind of research was done here in the United States and exported to the world. Now we look to 23 other countries for research ranging from general plant safety to severe accident models, waste management and advanced reactor design. If all goes well, we will soon add a 24th country to this official list of nuclear partners: the nation of Yugoslavia.

I had the opportunity this past summer to visit nuclear installations in several European countries, and I must say I was mightily impressed with the state of nuclear science in those countries, and a bit less impressed than before my trip with what we in the United States have done with our great nuclear power potential.

It is true we still have the largest nuclear power program in the world. It is true that, even with all our celebrated problems, we still have the largest nuclear construction program in the world. It is true that we continue to be a world leader in the research fields of computer code development, instrumentation, severe fuel damage, seismic research, and certain other nuclear frontiers.

And I believe economic and environmental necessity will very likely drive us toward a still larger nuclear enterprise in the United States over the next 15 years, as our economy continues to grow, as our electricity requirements grow with it, and as the dangers of acid rain, "greenhouse effects," and other drawbacks of fossil fuel use become ever more apparent.

But having said all that, I need not say that the United States -- alone among the nuclear nations of the world -- is taking five years to finish the last two percent of its latest nuclear plants, at costs 60 percent higher on average than the reactors of ten years ago.

The United States alone has come to experience the shocking spectacle of multi-billion dollar public works projects -- for that is what electric power plants really are -- standing idle or abandoned to rust in the field.

The United States alone among the countries with advanced nuclear programs has a plant reliability that hovers just under 60% while Britain, France, West Germany, Sweden, Finland, Canada, Switzerland and Japan have used similar technology and made it reliable up to 80% of the time.

In the United States, had we had a truly competitive utility industry, the diversity and resourcefulness of several tested and strong nuclear utilities could have been the engine that drove us and sustained us as leaders in applying our own technology. Instead, we alone have come to see dozens of reactors custom-built once by some 40-odd utilities, while in other countries a single reactor has been replicated quickly and efficiently time after time.

Let me hasten to add that there is plenty of blame to go around for the U.S. nuclear situation. We regulators must bear our share of responsibility, as must the Congress who contrived the regulatory and developmental framework for the nuclear enterprise, and as must the industry, some of whom were slow to recognize that nuclear power is serious, high-technology, space-age stuff, demanding of the very highest standards we have ever had to apply to any endeavor, whether in space, communications, or defense.

And as if all of this were not enough, the nuclear enterprise has often failed to convince a skeptical public that nuclear energy is the safe and environmentally sound alternative to fossil fuel that many believe it to be. A recent public opinion survey, for example, found that 80% of the people in the United States, and nearly as many in ten European countries, still believe it is possible, if unlikely, for a Hiroshima-type explosion to occur at an operating nuclear power plant.

That is the context in which regulators, researchers, vendors and utilities must work. Doing our job right is not enough. Being perceived by a profoundly skeptical public to be doing it right is at least equally important.

In the regulatory field in the U.S., completing three major pieces of unfinished business may allay the public's concerns, and enhance both the science and the commerce of nuclear power. These needed advances are standardized plant designs, resolution of unresolved safety and risk questions, and more assiduous quality assurance.

The time has long since come for the United States to join the rest of the nuclear world in approving standard plant designs. I do not believe we have to wait until every last word is in and every "global" issue is solved before we can say with confidence "this design is good enough, safe enough for adequate protection of public health and safety."

In fact we have built what could have been our first standard plant designs with the Combustion Engineering plants at Palo Verde in Arizona, the Westinghouse SNUPPS plants at Callaway in Missouri and Wolf Creek in Kansas, and the General Electric BWR-6 plant at Grand Gulf in Mississippi.

This is not to say that a standard plant, once approved, may not be improved. Quite the opposite is true; the standardized plant of today will never be the standardized plant of tomorrow. But as the NRC seeks to finalize its Severe Accident Policy considerations, it should expeditiously, if belatedly, be prepared to speak clearly on final design approvals for the standardized plants of the future.

I believe that each construction/operating permit issued for such a standardized plant should, on the day of its issue, carry with it the assurance that, short an urgent new discovery in reactor safety, the NRC believes that plant to be suitable for a standard licensed lifetime of nuclear power generation, without further modification.

But none of us should be afraid to recognize that we may even soon leave current light-water technology behind entirely, and move forward to something else. If today's standard plants are safe enough, then it could be that the challenge of the future lies in developing simpler, more passive, more economically attractive reactors.

The current-generation light-water reactor may well be reaching the point which the propeller-driven aircraft of the 1940's reached at the dawn of the jet age: they were proven, reliable, efficient, even elegant -- and they were obsolete. By the late 1940's, the propeller aircraft had gone as far as the laws of nature and the practice of engineering could take it, and the jet age was born of necessity.

It may be that one or more elements of the light-water technology borrowed from the naval nuclear propulsion program will be the propeller of the nuclear enterprise. The design features that made today's LWR attractive in the first place -- compactness, efficiency, relatively convenient and proven materials and systems technology, reliability, and low capital costs (at least in a properly executed construction program and a stable regulatory environment) -- may also define its limitations.

The use of a high power-density core for compactness requires very substantial external decay heat removal by forced convection. The use of water as a coolant, though highly efficient, requires replenishment, pressure control and often pressure maintenance. The use of high pressure and temperature, coupled with steel-based pressure boundary materials, creates problems with corrosion, pressurized thermal shock and other potential sources of severe accidents. These characteristics have led, in turn, to a vastly complex and cumbersome body of nuclear regulation in this country, jerry-rigged and back-fit to cover a seemingly endless variety of circumstances.

Some other countries which did not adopt the early light-water reactor as quickly or as fervently as we did are now showing us alternatives.

The PIUS reactor from Sweden and similar designs emerging here, though arguably LWR's, are almost completely passive in their response to transients and accidents. The Phenix reactor in France demonstrates that the use of water technology is not required for a large, reliable, and compact plant.

The AGR in Great Britain and the THTR in Germany demonstrate that a practical reactor can be built whose maximum credible accidents are not catastrophic. And the Canadian heavy-water reactor program has shown us that low power-density reactors can be economically competitive.

In light-water technology itself, we have learned a great deal from foreign research. We have learned from the French the value of standardization. The Germans have been engineering innovators in decay heat removal, vessel technology, containment design, and protection from sabotage. We have learned from our friends in Sweden, Japan, Finland, and elsewhere that LWR's can actually be as reliable as first envisioned. And we have learned from the British that fresh perspectives can further improve, even today, our own basic designs.

Still, there is the legitimate concern that the present generation LWR has gone about as far as it can go. And after years of neglect, the NRC has begun to concern itself more with safety research of advanced reactors--research designed to anticipate, not react to, the future. We have embarked on this program for a several reasons.

First, Congress told us to. With that for a first reason, we don't really need to look for a second or a third. But we're also moving toward more active involvement in advanced reactors because the public needs to know -- from a hard-eyed, independent, authoritative source -- whether the safety claims often made for new reactor systems have merit, whether they truly offer significant improvements over the current generation of reactors.

It is my opinion that when it comes to plant designs, we at the NRC ought to look at everything that comes our way and give each legitimate new reactor concept an appropriate evaluation. That means the NRC should be capable of a measured, fitting response to new reactor designs, whether HTGR, small modular liquid metal, large integral steam-cooled, or anything else you and others can imagine. If a back-of-the-envelope design is put on our table, we owe the public a back-of-the-napkin response. If a complete conceptual design is offered, we should provide a detailed response on the safety characteristics of that design.

In a draft regulatory policy for advanced reactors, now before the Commission, we've set forth the basic objectives for advanced reactor designs:

1) They should emphasize intrinsic or "built-in" safety features.

2) They should accommodate design basis events with a minimum of operator actions and equipment performance -- especially equipment subjected to severe environmental conditions.

3) They should minimize both the potential for, and the consequences of, severe accidents by designing reliability, redundancy, diversity and independence into the safety system, and they should minimize the number of challenges to those safety systems by providing a reliable balance of plant.

4) They should minimize the number of components that have to function to maintain safe shutdown conditions.

5) They should be less complex, so that safety precautions can be less complex and operator response more manageable in the event of "off-normal" conditions.

To summarize these objectives in a word, perhaps we should take our cue from the real estate business, where the first three criteria for selecting real estate, I was told when I bought my first house, are location, location and location. As we move into the 1990's, the first three demands on nuclear plant design should be simplify, simplify, and simplify.

And in the process, all of this may help simplify our effort to quantify risk and reach a definitive safety goal for nuclear reactors. A variety of safety goals are, in effect, on trial until next year here in the United States, when the NRC will decide which of these goals should be embodied in regulation.

For example, in preparing the goals in draft form, consideration was given to the inclusion of a quantitative containment performance goal. Recent research has given us a much better understanding of what qualities a reliable containment should possess.

Sixteen such qualities -- including ultimate strength, penetration reliability, venting capability and isolation, among others -- have been identified and arranged in a matrix, but no single number has been fixed which defines the containment's contribution to reducing risk.

But the number that defines risk will never be "zero," and the question then becomes "how safe is safe enough?" The decision of the Commission in this matter could have a profound effect in assessing the need for further backfitting, the priorities for resolving remaining unresolved safety issues, and the future of a number of research programs within the NRC.

Finding a way to assess risk quantitatively is something the NRC and its predecessor agency -- the Atomic Energy Commission -- have spent the better part of ten years studying. Our probabilistic risk assessment program has been trying to estimate the probability and consequences of a range of nuclear accidents by factoring such diverse considerations as engineering standards, geological surveys, reactor designs, containment safeguards, population exposure, previous experience, and perhaps a few educated guesses, into a mathematical model.

But while the science of risk assessment may be our province at the NRC, the art and science of risk acceptance is a profoundly political and social question, and I believe it is the Congress of the United States, with its finely honed sense of what the public will accept -- rather than the Nuclear Regulatory Commission -- which should provide the leadership on this most emotional and difficult question.

That congressional leadership has not yet been forthcoming, and in its absence the NRC has undertaken to set its own safety goal. We're considering the establishment of a quantitative increment of risk; we're looking at cost-benefit analysis; we're exploring various thresholds for intervention, in lieu of a single quantitative goal; and we're studying what amount of "life shortening" risk society may find acceptable.

So for all of these reasons -- technical, social, and political -- we on the Commission are the first to acknowledge that it would be much better of the people's representatives in Congress gave us more definitive marching orders about how safe is safe enough, and about how that "safe enough" goal should be quantified.

Even though we haven't yet reached the final precise "magic number" which quantifies an acceptable safety goal, the NRC already seems to operate with a rough internal sense of what we consider adequate safety for present generation reactors, and what standard we believe future generation reactors should meet. And in the absence of a definitive declaration on this subject, we owe the public -- not to mention groups like this, from whose collective wisdom the shape of the future will emerge -- at least some idea of what our working definition of "safe enough" is. That is a case I intend to press with my colleagues on the Commission now.

But despite all our efforts, the public often perceives design concepts, severe accident policy, and safety goals as impenetrable and more than a little unsettling. So I will make one plea for an eminently practical risk-reducer, and then I will close.

Let me begin by confessing that I continue to harbor the suspicion that the United States Government spends considerably more of its resources inspecting $2 chickens than it spends inspecting the construction of $2 billion nuclear powerplants. Not long ago our staff proudly told us that 30,000 hours of on-site inspection had been carried out by the NRC prior to licensing a new powerplant for operation. Well, that seemed impressive until I pulled out a pencil and calculated that over the 10-year construction period that averaged out to a little over 1.5 on-site inspectors (and a $2,000,000 commitment at best) from the federal government in the form of the NRC.

In contrast the West Germans have a dozen or more independent inspectors at each site throughout construction -- experts from the century-old guardian of German quality assurance, the Technische Ueberwachungs Verein, or TUV.

We, of course, have nothing that resembles a third-party TUV-style quality-assurance effort in the United States. But the Commission is currently grappling, under mandate from the Congress, with the question of how we may go about assuring quality of construction so that never again will we see the catastrophic spectacle of multi-billion dollar public works projects rusting in the field.

Various suggestions have been offered, including adoption of the designated representative concept employed by the Federal Aeronautical Administration. Whatever mechanisms for improved quality assurance might finally emerge, whether a USTUV or something else, I suspect the American people will demand a truly independent assurance of quality for insurance of their investment before the next generation of reactors gets off the drawing boards and under construction.

I also believe that the United States and its nuclear research partners must engage in a great deal more of the kind of information and experience exchange that you have had here this week. Every nation's research budget is under the knife these days, and cooperation and collaboration of effort are more important than ever.

While I was in Europe last summer, my hosts in each country offered the gentle but genuine complaint that the only Americans who came to see them were people like me -- Commissioner types who blow into town one day for a whirlwind of meetings and blow out again the next day. They asked for more of the mid-level technical people and the plant managers who could spend a week with them, learning and teaching.

Your meeting this week is an important step in the direction of greater international cooperation, and while this meeting has always been in the United States, maybe it's time for another country to host it. That would be an appropriate symbol. America's nuclear energy monopoly is long since past. The energy challenge is a global one, and some of the best work toward meeting that challenge is being done far from these shores.

I look forward to continuing and strengthening the partnerships we have developed these past twelve years, and I believe that together we can make the next twelve years and beyond a time of excitement and excellence in nuclear research.

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RESEARCH CHALLENGES

Twelfth Water Reactor Safety Research
Information Meeting

October 26,1984
Gaithersburg, Maryland

Remarks by Denwood F. Ross, Deputy Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

I would like to attempt to make a very brief summary of some of the important conclusions that have come from the sessions so far this week. I note that the attendance and the participation this year seem to be as high as ever. Approximately 750 people, not only from the United States but also from 16 other countries, have participated at one time or another this week. As you know, we started with the opening remarks from Dr. Miyanaga of JAERI and Commissioner Asselstine and then ended with closing remarks from Commissioner Bernthal. He noted that we had 28 sessions covering all areas of reactor safety research. I had asked the session chairmen to take notes during the individual sessions of what appeared to be the important challenges of the future, the work not yet done, to see if there could be some consensus on where research around the world should be heading. I note that, in addition to these formal publicized 28 sessions, there have been a number of workshops, secondary meetings, and ad hoc discussions which were also productive. In Monday's session, we had some discussion on the subject of thermal hydraulic and integral systems tests. It was noted that, in the boiling water reactor facility in California called FIST, the testing has been completed and no further testing is planned for this facility; no further needs were identified. There were additional discussions on the Semiscale facility, and testing should be finished in this facility (as it is presently built) in about 2 years. The NRC is currently building, in partnership with others, a test facility in the State of Ohio to test the B&W integral system design. This facility should be completed in 1986, and the testing will possibly extend through 1987. The general discussions on these integral system tests show that the NRC finds itself with two questions that are seemingly paradoxical. The first question is: can we justify maintaining and operating these expensive facilities, either in whole or in partnership with domestic friends, once their missions are complete? On the other hand, can we justify <u>not</u> having such facilities, and the expertise associated with these facilities, for future needs. Mr. Minogue in his Monday speech noted that we were indeed considering a new generation of integral system testing beyond the FIST, Semiscale, and MIST family. If we are to proceed in those areas, we will have to decide soon to develop some mission requirements and planning requirements and then decide where and how we would fund such an operation. It seems obvious that other countries have similar challenges in the integral system testing area. This emphasizes the fact that joint research planning is now a must for these more expensive projects.

A close ally of the integral system testing engineer is the separate effects engineer. There were a number of papers in the separate effects area covering two-phase flow and thermal hydraulic research. There seems to be an increased emphasis here on understanding the fundamental physics behind thermal hydraulic phenomena, especially those associated with behavior beyond the design basis. Such basic parameters as the interfacial shear between the phases, the condensation model, or the flow blockage heat transfer model all represent separate effects that need to be studied in greater detail if we are really going to have best-estimate prediction models.

In the area of mechanical engineering, the session on Monday noted several points as useful for future research. The piping supports vary from design to design and from country to country. If there is going to be a change in policy regarding the piping support, should there be more supports or less supports, and what should be their nature? What then is the tradeoff as you change from piping support policy for piping reliability and what is the new likelihood, say, of a large break for a given plant in a given year? If you alter the piping support policy what does this due to the nozzle loads as you get different break geometries and how does the nozzle load affect the likelihood of a small break propagating into a bigger break? A closely allied question has to do with a venture that is taking place in several countries, including the United States: shall the double-ended so-called guillotine pipe break be removed as a design basis for certain loads such as piping support or thrust loads? If you postulate that you know enough about piping reliability to eliminate the double-ended break, what is the next failure mode of the system? How will this affect the future design?

On Tuesday, one of the early sessions had to do with a containment systems research, called severe accident sequence analysis or the SASA program. There are several future needs of the SASA program. Some of the computer codes (I am emphasizing here the TRAC code) appear to be credible up to the point that the core geometry is lost; beyond that the results are somewhat tentative. There is an effort to couple TRAC with some of the core degradation codes, for example, the melt progression code named MELPROG. Development and verification will be quite difficult especially in the MELPROG part because there is no known experiment that will adequately verify all the aspects of the melt progression model. The likelihood of core melt with the reactor system at high pressure - by high pressure I mean perhaps 50 to 150 bars - is uncertain. What happens if you have a meltthrough and than have dispersal of the melt at high pressures? This appears to be an additional containment load factor sometimes called the direct heating mode. What is the containment response to this direct heating? It was concluded that more evidence in this area is needed. In general, the chemistry of fission product release and transport needs a lot of work if we are going to proceed to better estimates in this area. There seemed to be a general agreement that improvements in SASA during the last 2 years could be useful in guiding the operator to actions he could take while a core degradation was in process.

In the containment system research area, there was a general feeling that more work is needed in the core melt/concrete or the hot solid concrete interaction area. There are many programs under way. It is expected that significant results will be obtained in the next 1 to 2 years. Some studies have been completed showing that there are some differences between the hot UO_2 interactions with concrete and those of the usual simulant, thermite. The containment systems research efforts on aerosols indicate that more mechanistic calculation of particle size is needed. There are experimental programs involving aerosol generators in the nuclear safety pilot plant at Oak Ridge, Tennessee; the LACE project in the State of Washington; and the DEMONA project in Germany. It appeared of interest to the people at this session to do some common tests for all these facilities to see if we could get a more interrelated data base.

In another session on severe accident sequence analysis it was pointed out that there are several codes that have been used for the same accident, maybe indeed for the same reactor. These competing codes, in particular RELAP and TRAC, should be evaluated with respect to accuracy and running time. similar comment would apply to the fuel degradation models and fuel melt models. There is a strong relationship between the SASA program and the PRA program, (the so-called risk family) in that the SASA codes tend to be more detailed and the risk codes tend to be more fast running. It was concluded that intercomparison of these two techniques should take place. Two comments from a panel session on international programs in thermal hydraulics indicated that we need better modeling and predictive capability for our thermal hydraulic codes in the postcritical heat flux area. In particular, there were some statements that the TRAC/PF1 needed improvement in the post-CHF area. There are comparisons of the TRAC code as it applied to the rewet phenomena in one of the OECD LOFT experiments. Also, and this is especially true for smaller breaks, the codes do not predict phase separation well. It is important to know if you have a loss of coolant at the top of a pipe, the middle of the pipe, or the bottom of the pipe because the depressurization rate, of course, is strongly dependent on the density of the existing fluid.

An entirely different topic on Tuesday was seismic research. There seem to be several major challenges in seismic hazard studies, seismic PRA studies, and planning to better quantify the seismic risk. The comments were divided into three parts. The first is the seismic hazard or the challenge from the geology. As far as the United States is concerned, we are currently more interested in reassessing the Eastern United States sites. There is a specific challenge especially as it might apply to a plant that is located on soil. Second, it was noted that more data are needed on fragility curves. The panel session noted that a more definitive determination of failure modes in the inelastic range is needed, in particular for piping. The third point made in the panel session was that we need better methods in systems analysis. Some points were also noted on seismically induced relay chatter and circuit breaker trip and how these would interact with the systems analysis of seismic risk.

On Wednesday there was a session on fuel system research. Here there seemed to be a consensus that the programs were generally mature. Papers covered the internationally used MATPRO code, Material Properties code, and the FRAPCON-2 steady-state fuel rod behavior code. The panel concluded that the PCI failure mode in fact represented little threat to the public health. A summary paper on the TMI-2 core examination showed that many of the fission products tend to be retained by the primary system, i.e., deposition and retention on metal surfaces.

Most countries in the nuclear business today are working on accident source term reassessment. The session on this subject noted that a number of issues have been identified by the NRC reassessment for the next 2 years, some challenges yet to be met (to give the final best estimate) would include: fuel melting and slump progression, recirculation flow patterns in the vessel as these lead to the deposition models, the in-vessel release of fission products; for example the silver aerosol; the reevolution of fission products from the reactor coolant system surfaces, and, probably most important, the containment performance in a severe accident: when does it start to leak, how much does it leak, and, eventually, what is the catastrophic failure mode if there is one?

A topic that has been of interest in several countries is pressurized thermal shock. Recently the NRC's committee for the review of generic requirements approved for transmittal to higher management a final rule on pressurized thermal shock. There are still some uncertainties that are worth research in the pressurized thermal shock area. These could give a more realistic picture and, when more detailed analyses of PTS by individual utilities come in, a more realistic assessment could be made. Quantification of 3-D effects in the downcomer is needed because the downcomer fluid provides the mechanical challenge to the vessel wall. In some cases, there is asymmetry between the loops. If you have a steam line break, one steam generator blows down, you tend to have asymmetric fluid temperatures in the downcomer. If you want to quantify this accurately, you will need a 3-dimensional code. If you want to verify that code, you will need some data. Data are being gathered, and it was agreed this was a good idea. For some of the sequences, if you inject high pressure injection into one loop, you can also get an asymmetric flow pattern, and additional studies are being performed by the NRC in cooperation with Finland to give a better estimate code here. Some of the thermal hydraulic work being conducted in model studies at the University of Purdue are also providing better estimates of the downcomer flow patterns. It was agreed that, with respect to pressurized thermal shock, some work in the future to give more precision would concern the number and location of the preexisting cracks in the clad vessels: how many flaws exist, where are they, how deep are they? In some instances the response to PTS involves the operator. Some questions there are what will the operator do, what must he not do, and how long will it take? These also could be used to quantify the likelihood of an accurate response to a PTS event.

Also on Wednesday, were discussions on the pressure vessel. There could be what is called a life extension mode in which some of the existing power plants would be qualified beyond their intended service life. If you want to go beyond say the so-called 40-year life and you have to requalify the system to get an extension of life, there could be a need for more data on large-scale testing on the postannealing properties if indeed the vessel had to be annealed to extend its life. If you are depending on annealing for radiation defects, would the thermal aging also anneal out? You might need more data on the as-manufactured defects in the vessel. Perhaps it might be useful to get some salvage value out of one of these vessels from a canceled plant to do some nondestructive as well as destructive assaying of the as-manufactured vessel flaws.

On Thursday, in the metallurgy and steam generator session, there was a conclusion that we should be continuing work on aging toughness of cast stainless steel. These would be studies at temperature on materials used for piping and pump casings. Also there appeared to be some interested on the aging of ferritic materials at operating temperature. These would, of course, apply both to piping and vessel materials. There were some discussions on one of the favorite topics to the PRA and severe accident research program people on steam explosions. There is certainly at present at least a mild difference of opinion on the possible significance to reactor safety of steam explosions. What do we need to do? What is the threat to reactor containment of a steam explosion, and what further research is needed to reduce the uncertainty here? This is one area where I believe the experts have yet to agree; this may point out the need for more sessions. There may need to be special sessions on topics where there seems to be a lack of consensus among the experts. There were discussions on hydrogen behavior. Again there is not a consensus among the experts: have we done all of the work on hydrogen behavior, both combustion and detonation, that we need to do? The bringing together of the experts here is helpful. I believe we can conclude there is an international consensus that we have not yet done enough; what remains to be done?

On Thursday, we had sessions on human factors. There were comments that additional human factors research would be needed in three areas. In the maintenance area, how do you qualify human errors? What is the measure of an effective maintenance program? As materials and components in systems start to age, what is the role of preventive maintenance in producing a reliable component? Do you need to study in detail the management and organization of a utility with respect to an effective maintenance program? What are the criteria and standards? And what is the interface between maintenance and operation? All seem to be in agreement that these are areas amenable to research. The second area involves the reactor operator, who is counted on to manage a potential accident. Can he manage it? What is the true response of the man side of the man-machine interface? There is a lot of work going on. There seemed to be an agreement that it is not yet done. In the third area, there were suggestions that an operator performance model, both computer simulation and analysis, is needed to better quantify the likelihood of human reliability as an operator.

There is a particular research program involving three countries represented here today: the 2-D 3-D research program. The theme of this thermal hydraulic research program, which was discussed Thursday is scale. These facilities, in particular the one in Germany, are large. The German facility is a full-scale thermal hydraulic test facility. It is expected that these facilities in Japan and Germany augmented by the analysis and instruments supplied from the United States would help or perhaps would completely solve such issues in emergency core cooling as steam binding, emergency core cooling bypass, and distribution of the emergency core coolant within the vessel, especially within the core.

It is kind of hard to summarize Friday, because the Friday sessions haven't taken place. We did get an opinion from the session chairman on the reliability session as to what he hoped would be the conclusions of the session yet to take place. He is hoping that there is a consensus that reliability techniques can and should be applied more to the nuclear industry. This would be done by surveying what has happened in commercial ventures other than nuclear to determine if we can learn from such things as the aerospace industry about reliability. There is a general feeling that, yes, you can. The question now is how do we get on with doing this? This would certainly require some industry cooperation and participation.

Also yet to be discussed today is the need for additional research on consequence modeling. Offsite consequence modeling can require two additional research projects. One would be to reduce the uncertainties in current models and the other would be to try to model phenomena that are not taken into account today, such as a deposition from a plume during its transport, especially a wet plume that one might have, a phenomenon sometimes called spontaneous rain (a deposition of major amounts of fission products in the vicinity of the power plant, during the accident, as contrasted with transport down wind). This could produce a major change in consequence estimates, and it seemed to be an area certainly amenable to research and certainly might have a large payoff.

But in conclusion, I think we should consider that we have had a fairly useful session. In particular the international sharing is noteworthy. I think we need not only to continue to emphasize in these sessions what has been done, which is what we usually emphasize, but also to give more attention to what needs to be done in the future especially when we are trying to point to a best-estimate characterization of reactor risk. We had a meeting last month in Karlsruhe, Germany, the ANS/ENS conference. During the closing panel session there, we emphasized the nature of international research and developing a theme internationally of what remained to be done. I think these sessions provide the needed input on specifics to cement the idea that collectively we can get the job done.

SIMULATOR EXPERIMENTS: EFFECTS OF NPP OPERATOR EXPERIENCE ON PERFORMANCE

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ABSTRACT

Experiments are being conducted on Nuclear Power Plant (NPP) Control Room training simulators by the Oak Ridge National Laboratory, its subcontractor, General Physics Corporation, and participating utilities. The experiments are sponsored by the Nuclear Regulatory Commission's (NRC) Human Factors and Safeguards Branch, Division of Risk Analysis and Operations, and are a continuation of prior research using simulators, supported by field data collection, to provide a technical basis for NRC human factors regulatory issues concerned with the operational safety of nuclear power plants.

During the FY83 research, a simulator experiment was conducted at the control room simulator for a GE Boiling Water Reactor (BWR) NPP. The research subjects were licensed operators undergoing requalification training and shift technical advisors (STAs). This experiment was designed to investigate the effects of (a) senior reactor operator (SRO) experience, (b) operating crew augmentation with an STA and (c) practice, as a crew, upon crew and individual operator performance, in response to anticipated plant transients.

Sixteen two-man crews of licensed operators were employed in a 2 x 2 factorial design. The SROs leading the crews were split into "high" and "low" experience groups on the basis of their years of experience as an SRO. One half of the high- and low-SRO experience groups were assisted by an STA. The crews responded to four simulated plant casualties. A five-variable set of content-referenced performance measures was derived from task analyses of the procedurally correct responses to the four casualties. System parameters and

control manipulations were recorded by the computer controlling the simulator. Data on communications and procedure use were obtained from analysis of videotapes of the exercises. Questionaires were used to collect subject biographical information and data on subjective workload during each simulated casualty.

For four of the five performance measures, no significant differences were found between groups led by "high" (25-114 months) and "low" (1-17 months as an SRO) experience SROs. However, crews led by "low" experience SROs tended to have significantly shorter task performance times than crews led by "high" experience SROs. The presence of the STA had no significant effect on overall team performance in responding to the four simulated casualties.

The FY84 experiments are a partial replication and extension of the FY83 experiment, but with PWR operators and simulator. Twenty-seven crews (24 three-man crews of licensed operators and three groups of pre-license trainees, plus STAs augmenting 12 of the crews) will respond to five operating sequences representing a range of difficulty from routine to very severe accidents. Results of the FY84 experiments will be available by March 1985.

INTRODUCTION

Since February 1983 the Oak Ridge National Laboratory (ORNL) has been conducting research on Nuclear Power Plant (NPP) control room operators' performance. This research is being performed in plant training simulators, using controlled experimental designs and performance measurements. The current project is an outgrowth and continuation of the Nuclear Regulatory Commission's (NRC) research project on Safety Related Operator Actions (SROA) and incorporates the methods and procedures developed under the SROA project[1] and the NRC Crew Task Analysis project[2]. The Simulator Experiments project provides operator performance data from simulators, supported by field data collections, to support a technical basis for NRC human factors regulatory issues concerned with the operational safety of nuclear power plants.

NRC licensees voluntarily participate in these experiments, contributing the time of their staffs (e.g., licensed operators, training specialists) and the use of their simulators. Each experiment consists of a series of predefined exercises to which operating crews respond in a simulator. These exercises (or "operating sequences") are developed from a comprehensive system/task analysis. Automated data recording, observation, videotaping, and self-report questionnaires are used to obtain individual and crew performance measures for each simulator exercise.

This paper reports on the methods, procedures, and results of the FY83 experiment and the plans for the FY84 experiment.

FY 1983 Experiment

The initial experiment was conducted at the control room simulator for a GE Boiling Water Reactor (BWR), from October through December of 1983. The purpose of this experiment was to investigate the contribution of senior reactor operator (SRO) experience, the presence or absence of a Shift Technical Advisor (STA), and practice as an STA-augmented crew to operating crew response to four anticipated plant transient exercises. The specific questions addressed by this experiment were:

a. Do operating crews led by more experienced SROs perform better than operating crews led by relatively less experienced SROs?

b. Do operating crews augmented by an STA perform better and/or experience lower perceived workload than do operating crews without STAs?

c. Is the performance advantage (if any) conferred by the assistance of an STA immediate, or is practice as a team required before any advantage is realized?

METHOD

Design

Sixteen two-man crews of licensed operators (an RO and an SRO) were employed in a 2 x 2 factorial design in which the factors were SRO experience and the presence or absence of an STA. The SROs leading the crews were split into "high" and "low" experience groups on the basis of their years of experience as an SRO. One half of the high- and low-SRO experience groups were assisted by an STA or an SRO acting as an STA. This resulted in four groups of four crews each:

a. high-experience SRO + RO (crews 1-4)

b. low-experience SRO + RO (crews 5-8)

c. high-experience SRO + RO + STA (crews 9-12)

d. low-experience SRO + RO + STA (crews 13-16)

The effect of practice as a crew on the performance of the STA-assisted groups was to be determined by analysis of changes in performance between the first and fourth exercises.

Subjects

Sixteen licensed ROs, nineteen licensed SROs, and five STAs served as the subjects. The design of the experiment called for eight STAs. However, three of the STAs were unavailable during the last month of data collection due to a refueling evolution, and licensed SROs were substituted for them.

The ROs and SROs were split into high and low experience groups on the basis of how long they had been licensed as an RO or SRO. The experience of the participating operator is shown in Table 1.

TABLE 1

Months Since License for SROs and ROs

		Months Licensed		
License	Experience	$\bar{X}$	SD.	Range
SRO	high	94	23	42-114
SRO	low	7	5	1-17
RO	high	46	14	31-66
RO	low	13	7	1-19

Experience as an SRO (in a supervisory position in the control room) ranged from 42 to 114 months for members of the "high" experience groups and from 1 to 17 months for the "low" experience groups, but all SROs in the "low" group had at least 3 years of experience as an RO. An attempt was made to balance the groups for RO experience, with two of the crews in each group having a "high"-experience RO and two a "low"-experience RO (this division for the ROs falling at two years), but was not entirely successful: group "d" crews had three high- and only one low-experience RO.

Exercises

Four simulated plant casualties or operating sequences were used:

a. Anticipated Transient Without Scram (ATWS) following turbine trip (the Reactor Protection System may be deenergized to scram the reactor).

b. Turbine Trip (TRIP)

c. Loss of Feedwater (LOFW) caused by failure of a booster pump.

d. Safety/Relief Valves (2) fail open (SRVF), followed by uncontrollable depressurization.

Procedure

The experiment was conducted in a plant-referenced training simulator located at the site of the plant, which was a two-unit BWR. The utility very generously allowed four hours of simulator training time on the first day of the operators' quarterly requalification training for the experiment. Each session began with a briefing of the participants, followed by instructions and scale development for the Subjective Workload Assessment Technique (SWAT) workload rating forms[3,4], followed by performance of the four exercises for which data were collected. The order of presentation was balanced across teams by the use of randomized Latin squares. The four operating sequences were set up by the instructor, who followed a protocol supplied by the experimenters.

System parameters and control manipulations were recorded by the computer controlling the simulator[5]. Data on communications and procedure use were obtained from analysis of videotapes of the exercises. Questionnaires were used to collect subject biographical information and data on subjective workload during each simulated casualty.

ANALYSIS

Two sets of content-referenced performance measures were derived from descriptions of the correct responses to the simulated casualties obtained from task analyses of the operating sequences. The task analyses were conducted by the methods employed by the NRC Crew Task Analysis project[2] modified by ORNL to identify and document the performance required of the operator in order to meet the system's requirements. Performance criteria were derived from the task analysis data and additional information from operating and administrative procedures that documented preconditions and limits associated with the actions listed by the task analysis (e.g., RCIC - reactor core isolation cooling - turbine speed should be reduced to less than 2000 rpm prior to tripping the turbine). Performance of each task was described by five measures:

a. Whether the task was initiated (Init) scored as 1 or 0;

b. The percentage of task elements (Elem) performed correctly;

c. The percentage of preconditions or limits (P/L) complied with;

d. Whether task success criteria (Succ) were met, scored as 1 or 0;

e. Time elapsed from the appearance of the cue to initiate the task until the task was completed, in seconds.

A second set of measures was based on control of system parameters; examples are time-out-of-band and RMS error for reactor water level and pressure. Preliminary analysis indicated that scores on these measures were not systematically related to the major variables investigated in the experiment.

Data on the performance of the individual elements comprising each subtask of each operating sequence were derived from the records of control actions and parameter values produced by the simulator's computer and analysis of the videotapes of each exercise. The five measures of task performance were averaged (task times were first standardized) across all tasks in the operating sequence to produce a single 5-variable set of task performance scores for each crew for each exercise.

RESULTS

Crew Performance

To compensate for the fact that we were unable to balance the four experimental groups with respect to the ROs' experience, the performance scores for the 16 teams were analyzed by means of a multivariate analysis of covariance (MANCOVA) with the ROs' months of experience as licensed operator as the covariate. The analysis was performed by means of the multivariate analysis program of the BMD-P statistical software package. The MANCOVA had two between-subjects factors, "SRO experience level (high or low)" and "presence of STA," and one within-subjects factor, "exercises." The averages on which the tests for main effects were made are given in Table 2.

TABLE 2
Comparisons Corresponding to Main Effects in the Analysis of Crew Performance

	Measure				
Effect	Init	Elem	P/L	Succ	Time[a]
SRO Experience:					
High	.849	.861	.751	.906	+.14
Low	.855	.863	.779	.905	-.13
STA:					
Absent	.841	.856	.772	.901	+.02
Present	.864	.868	.758	.911	-.01

a. The time measured is reported as deviation units (Z-scores).

The test for the significance of the covariate (months since RO license awarded) was significant for the set of five measures ($F(5,7) = 4.13$, $p < .05$). The effect of the covariate was not significant for any individual measure. However, examination of the data showed that the crews with the more experienced ROs tended to score slightly higher on four of the five measures than did crews with less experienced ROs.

For the five measures as a set, the multivariate F for SRO experience level was significant: $F(5,7) = 4.43$, $p < .05$. Examination of the data presented in Table 2 reveals that there are small differences between the SRO-high and SRO-low groups on all five measures, but that these differences run counter to expectation: the SRO-low groups scored higher on four of the five measures than the SRO-high groups.

Univariate ANCOVAs having the same factors as the MANCOVA were performed for each of the five task performance variables. In these analyses, the difference between the SRO-low and SRO-high groups was statistically significant for only one measure: the SRO-low crews tended to respond more rapidly than did the SRO-high crews $F(1,11) = 8.14$, $p < .05$.

Effects of Presence of STA

There were no significant differences in overall performance due to the presence of an STA to assist the crew: $F(5,7) = 1.01$. Inspection of Table 2 shows that the STA-assisted crews tended to receive higher scores on four of the five measures than did the crews without an STA, but none of the differences are statistically significant. The interaction between SRO experience and the presence or absence of an STA was also not significant: $F(5,7) = 1.36$.

There were significant differences among the four operating sequences for the set of measures as a whole: multiple $F(15,89) = 4.46$, $p < .001$. These may be traced to the characteristics of particular tasks within each sequence.

Effects of Practice

The analysis to address the third objective of the experiment, determination of the effects of practice as a crew on the performance of the STA and no-STA groups, examined the change in the task performance measures from the first to the fourth exercises performed during the experiment.

To render potentially dissimilar exercises equivalent for the purposes of the analysis, the performance measures for each operating sequence were standardized (expressed as a Z-score relative to other scores on each variable for the sequence) to allow aggregation across exercises. The standardized scores were analyzed by means of a mixed factor MANCOVA in which the between-subjects factors were "SRO experience level" and "presence of STA," the within-subjects factor was the order (first, second, etc.) in which the exercise was presented, and the covariate was again the ROs experience.

The main effects of the between-subjects factors paralleled those in the first analysis. For the set of five measures, the effect for order of presentation was not significant: $F(15,89) < 1$. However, the supplemental ANCOVAs for the individual performance measures indicated significant differences due to order of presentation for the time variable ($F(3,36) = 5.20$, $p < .01$). Average task performance time decreased steadily from the first ($\bar{X} = +.19$) to the fourth ($\bar{X} = -.14$) exercise. This probably represents a "warm-up" effect. The reduction in task time over the four exercises was somewhat greater for the crews with the high-experience SROs than for the crews with low-experience SROs, but the interaction was not significant.

The effect of practice working with the STA (as opposed to an overall warm-up effect) should be reflected by an interaction of the "order" and "presence of STA" factors. For the set of performance measures as a whole, the interaction was not significant: $F(15,89) < 1$. For the task time measure, this interaction was marginally significant: $F(3,36) = 2.58$, $p < .10$. Crews assisted by an STA tended to have shorter task times for the first exercise of the day than did crews without an STA, but this relation was reversed for the fourth exercise.

Perceived Workload

Operators and STAs reported their perceived workload during the performance of each operating sequence at the conclusion of each exercise. The major findings were:

a. There were no reliable differences between the perceived workloads of SROs (or ROs) in the STA and No-STA groups.

b. There were significant differences in workload for SROs, ROs, and STAs across the four operating sequences. ATWS was highest, followed by SRVF, followed by LOFW and TRIP, which did not differ significantly. Nine of 16 SROs and six of 16 ROs reported maximum or near-maximum workloads (85 or higher on a scale of 0 - 100) for the ATWS sequences.

c. The self-reported workloads of STAs were lower than the workloads of ROs and SROs for all sequences. The ATWS sequence was the only one for which the STA workload averaged above 50. This suggests greater STA involvement in this sequence, which was the only non-routine casualty simulated.

DISCUSSION

The experiment employed a basic two-man crew consisting of an SRO and an RO. This provided a somewhat unusual situation for the participants, as the normal operating crew consisted of three men, an SRO and two ROs. Two-man crews were employed for two reasons: there were not enough ROs at the participating plant to form 16 three-man crews, and the use of two men instead of three would make the SRO's job more difficult, as he would be required to perform some of the actions normally performed by the second RO. In spite of the increased workload due to the two-man crew, all crews performed adequately and were able to restore the simulated plant to a stable and safe condition within the time allotted to each exercise.

It was anticipated that the two-man crew would tend to accentuate experience-related differences in performance, and also to provide more opportunity for the man in the STA role to contribute to crew performance.

The major experience-related difference in performance was that the crews with the less experienced SROs executed required tasks more rapidly. Within the range of SRO experience represented in the crews who participated in this study, the experience level of the SRO directing the crew had little effect upon the other aspects of performance measured.

The generality of this finding is constrained by the fact that SRO experience was distributed in the available subject population in a way that was less than ideal for the purposes of the experiment. Even the less experienced SROs in this study had at least 40 months of operating experience. The absence of significant performance differences as a function of SRO experience may be an artifact of the population available for the experiment: proficiency may reach near-asymptotic levels for licensed operators beyond a given level of experience. If the question of "the minimum level of experience" required for SROs is to be examined empirically, it is desirable to include the range of experience from newly licensed operators through the levels represented in this study.

Interpretation of the data with regard to the apparent absence of measurable benefits conferred by the STA is also problematic. Since procedures were available for responding to all of the simulated casualties, the engineering expertise of the STA was not needed. Thus it may be argued that the study did not provide a fair test of the value of the STA (see Ref. 6 for a discussion of the contribution of the STA during transients). However, the two-man crew situation should have allowed the STA to contribute to crew performance by assisting the SRO to monitor plant parameters and maintain an overview of the situation as it unfolded, verifying that procedurally required actions had been performed or suggesting them if they had not, and initiating required notifications.

Although performance was evaluated in terms of timely and correct execution of actions that were usually called out in the procedures, the above contributions should have been reflected in many of the measures. A tendency for the STA-assisted groups to score higher on four of the five measures was in fact observed, but the differences obtained were not large enough to achieve statistical significance.

Because of the limitations discussed above, the results of this experiment must be considered as suggestive rather than conclusive.

FY 1984 Experiments

The experiments planned for 1984 are a partial replication and extension of the FY 1983 experiment, with PWR operators in a PWR simulator. In addition to comparing SRO experience with performance and measuring the effect of STAs on crew performance, the FY84 experiments will also:

1. ensure the reliability and generalizability of the findings of the 1983 experiment;

2. further develop and refine the performance measures developed for the FY 1983 experiment;

3. extend the range of operating sequences to include true emergency conditions (where the STA's contribution becomes more critical) in addition to the anticipated transients employed in the 1983 experiment;

4. expand the range of operator experience by including operator trainees; and

5. compare the performance of operator candidates undergoing pre-license qualification training to the performance of licensed operators undergoing regular requalification training.

Details of the experimental design depend upon the number and characteristics of the operators available and the degree of support (e.g., simulator time, operating sequences thought to have training value, etc.) offered by the participating utility. The utility supporting the 1984 experiment has agreed to make available 24 three man crews of licensed operators, 12 STAs, and three groups of pre-license trainees. In consultation with the utility, five operating sequences tentatively have been selected for the 1984 experiment:

1. the turbine-loading segment of a unit startup;

2. turbine trip during startup;

3. steam generator tube rupture at full power;

4. total loss of feedwater; and

5. main steam line rupture outside containment with a steam generator tube rupture and puncture of the refueling water storage tank (the primary source of reserve coolant).

These five sequences represent a range of difficulty from the routine (sequences 1 and 2) to a very severe accident (sequence 5). Task analyses have been performed and verified in the training simulator and data collection began in August 1984. Field data exist for three of the five events (No. 1, 2, and 4). Results of the FY84 experiments will be available by March 1985.

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Development of Methods for Nuclear Power Plant Personnel Qualifications and Training*

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ABSTRACT

The Nuclear Regulatory Commission (NRC) has proposed that additions and revisions should be made to Title 10 of the "Code of Federal Regulations," Parts 50 and 55, and to Regulatory Guides 1.8 and 1.149. Oak Ridge National Laboratory (ORNL) is developing methods and some aspects of the technical basis for the implementation and assessment of training programs, personnel qualifications, and simulation facilities to be designed in accordance with the proposed rule changes. The paper describes the three methodologies which were developed during the FY-1984 research. The three methodologies are: (1) a task sort procedure (TSORT); (2) a simulation facility evaluation methodology; and (3) a task analysis profiling system (TAPS).

The task sort procedure was developed to determine the training strategy which should be applied to a given task. It accomplishes this by sorting tasks into nine categories, each of which is defined along ten dimensions. TSORT provides rank-ordered preferences for task allocation between and within categories and has the ability to estimate a dollar loss incurred through failure to train on a task. The simulation facility evaluation methodology is being developed to certify simulation facilities for use in the simulator-based portion of the licensing examination. It is to be utilized during two phases of the life-cycle, initial simulator testing and recurrent evaluation. The initial testing phase is aimed at ensuring that the simulator provides an accurate representation of the reference plant, while recurrent evaluation is aimed at ensuring that the simulator continues to accurately represent the reference plant throughout the life of the simulator. The task analysis profiling system has been designed to support training research. It draws on artificial intelligence concepts of pattern matching to provide an automated task analysis of normal English descriptions of job behaviors. TAPS development consisted of creating a precise method for the definition of skills, knowledge, abilities, and attitudes (SKAA), and generating SKAA taxonomic elements. It systematically outputs skills, knowledge, attitudes, and abilities, and information associated with them.

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Introduction

The Nuclear Regulatory Commission (NRC) has proposed that additions and revisions should be made to Title 10 of the "Code of Federal Regulations," Parts 50 (Training and Qualifications) and 55 (Operating Tests), and to Regulatory Guides 1.8, "Personnel Qualifications and Training for Nuclear Power Plants," and 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations (proposed title change)." The revised training rules would require each nuclear power plant licensee and applicant for an operating license to establish and implement training programs that are derived from a systems approach to training (SAT) for civilian nuclear power plant (NPP) operators, supervisors, technicians, and other appropriate operating personnel. SAT is an orderly, iterative process in which analysis of the job to be performed provides needed information for key decisions about training. Changes to the qualification rules would make it necessary for NPP licensees to follow specific guidance on qualifications or to base qualification determinations on findings from a systematic analysis of prerequisite qualifications and job performance requirements. NRC rules on the operating test would require that it be administered in a plant walk-through, and in a simulation facility which could be the plant, a plant-referenced simulator, or another simulation device, alone or in combination. The simulation facility is to be evaluated, as to its appropriateness for the conduct of the operating test, by the facility licensee for each nuclear power unit.

The Human Factors and Safeguards Branch, Division of Risk Analysis and Operations, Office of Nuclear Regulatory Research (HFSB/DRAO/RES) is supporting Oak Ridge National Laboratory (ORNL) to develop methods and some aspects of the technical basis for the implementation and assessment of training programs, personnel qualifications, and simulation facilities to be designed in accordance with the proposed rule changes. This paper will describe the three methodologies which were developed during the FY-1984 research. Earlier efforts in the program are documented in Haas, Selby, Hanley, and Mercer (1983)[1] and Selby and Hensley (1984)[2]. The three methodologies reported here are: (1) a task analysis profiling system (TAPS); (2) a task sort procedure (TSORT); and (3) a simulation facility evaluation methodology. TAPS will be covered in detail, but the other two methodologies will be treated in only a cursory fashion since work on TSORT was presented at a previous Water Reactor Safety Information Meeting (Jorgensen, Haas, Selby, and Lowry, 1983),[3] and the simulation facility evaluation methodology is to be field tested and further refined during the current fiscal year.

A Task Sort Procedure

Background

During the implementation of a systems approach to training, a variety of analyses must be performed which require the subjective expertise of a training developer. One major analysis is the determination of where (i.e., in what setting, training category, etc.) individual job tasks should be trained and how they should be ranked relative to different instructional aids and approaches. Depending on the skill of the personnel making the decisions, the resulting allocation of tasks to training strategies may or may not be made properly. In SAT, the kinds of courseware developed, the media and methods used, and the types of student evaluations performed are directly influenced by the general training strategy. There is thus a "ripple effect" from poor decisions which have been made early in the process. For the NRC, faced with evaluating many different training programs, it thus becomes important to have an objective basis to determine whether industry selections are reasonable within the SAT framework.

Method Description

TSORT was developed to determine which training strategy should be applied to a given task. It accomplishes this by sorting tasks into nine categories: qualification, certification, and refresher training, candidate for more or less training, potential simulator or formal training task, and candidate for on-the-job training or for elimination from training. Each category is defined along ten dimensions:

1) skill acquisition difficulty
2) skill performance difficulty
3) immediate performance need
4) safety consequences
5) previous nuclear experience
6) normal operation performance
7) emergency operation performance
8) plant delay tolerance
9) regulatory requirement
10) economic consequences

TSORT provides rank-ordered preferences for task allocation between and within categories. The ability to estimate a dollar loss incurred through failure to train on a task is included. A realistic economic model was beyond the scope of the work, but could readily be incorporated. The sort procedure is programmed for an IBM personal computer, menu-driven, and fully interactive for both data entry and analyses.

A Simulation Facility Evaluation Methodology

Background

It has long been recognized that simulators provide great potential for training and testing people on many types of tasks, both in nuclear power generation and other technical training endeavors. Of particular relevance to NPP training/testing, simulators provide a mechanism for training operators how to effectively respond to off-normal conditions. Because of this reliance on simulators for training/testing important tasks, there is an increased danger associated with the simulator being responsible for improper training. If the simulator does not behave in the same manner as the actual power plant, then the operator may be mistrained.

Industry organizations and the NRC recognize this potential problem and have taken steps to ensure that simulator training is effective. They have developed guidelines which define the type of malfunction to be simulated and the quality of simulation required. However, very little guidance is available to determine the acceptability of the simulation.

Method Description

The simulation facility evaluation methodology is being developed to certify simulation facilities for use in the simulator-based portion of the licensing examination. It is to be utilized during two phases of the life-cycle, initial simulator evaluation and recurrent evaluation. Initial evaluation is to be performed when a simulator is acquired, in the case of new simulators, and as soon as is practical for existing simulators. This phase of evaluation is aimed at ensuring that the simulator provides an accurate representation of the reference plant. There are two components of initial simulator evaluation: fidelity assessment and a direct determination of the simulator's adequacy for operator testing. Recurrent evaluation is aimed at ensuring that the simulator continues to accurately represent the reference plant throughout the life of the simulator. It involves three components: monitoring reference plant changes, monitoring the simulator's hardware, and examining the data from actual plant transients as they occur.

A Task Analysis Profiling System

Background

Task analysis is generally a highly subjective process that draws on observations of job performers' behaviors and combines them with an analyst's expert knowledge of systems to produce a functionally useful set of skills, knowledge, abilities, and attitudes (SKAA). The procedure often winds up being an art rather than a science and, as a result, is subject to a variety of shortfalls characteristic of highly subjective procedures.

Because task analysis is used for a variety of research purposes including courseware development, entry level skill identification, performance standards development, and personnel selection, large variations in task analysis quality can be very costly in time and resources. Unanticipated costs often occur as a result of repeated site visits to extract missed information, correct erroneous assumptions, or modify incorrect courseware materials. The end result is growing pressure for a faster, more economical method to support training research.

The task analysis profiling system has been designed to remedy these problems. It draws on artificial intelligence concepts of pattern matching to provide an automated task analysis of normal English descriptions of job behaviors.

Method Development

SKAA Definition. To support the automation of task analysis, a much more precise method for the definition of SKAAs had to be created. For example, a defintion of a human ability such as "perceptual speed" has generally relied upon text descriptions and the opinion of a task analyst such as the following:

> "The ability to compare sensory patterns quickly in order to determine identity or similarity."

Although appearing easy to use, such a general definition can lead to a great deal of disagreement over what constitutes a perceptual speed instance. For example, it may be understood that perceptual speed is a visual ability combined with a cognitive activity of recognition or recall. It is not clear, however, if the "sensory patterns" could also refer to other senses, e.g., auditory recognition of Morse code strings, or tactual recognition by a pilot of changes in g-forces. Thus, from the standpoint of an automated tool to identify perceptual speed, a more precise method of definition is needed. Mallamad, Levin, and Fleishman (1980)[4] recognized the same problem and proceduralized some ability definitions through a series of question/answer flow charts that eventually led to the identification of individual abilities. Unfortunately, such systems and efforts to automate them (e.g., Rossmeissel, Tillman, and Best, 1982),[5] place a tremendous resource demand upon a user since each task must be scanned for each individual ability through a separate question/answer path. To apply such an approach to field analysis of SKAAs would quickly produce massive resource demands on the analyst that would outweigh the benefits of the faster, albeit "noisier" subjective approach.

TAPS has taken another approach in that it recognizes from the onset that resource demands on the user are a critical component in the ultimate implementability of a training research tool. The key to the success of the approach lies in its ability to define SKAAs in a flexible manner capable of accepting many different potential sentence variations of the same underlying idea. Thus, "rapidly spotting a change in a temperature gauge" or "detecting a panel meter deviation in less than ten seconds" must both be recognized as an instance of perceptual speed by the definitional rule.

TAPS gains this flexibility through an approach analogous to choosing dinner items from a Chinese menu. In a typical Chinese dinner, an acceptable "meal" is defined as picking one item from column A, one from column B, and one from column C:

		A	B	C
		Item 1	Item 1	Item 1
		Item 2	Item 2	Item 2
"meal"	=	Item 3		Item 3
		Item 4		

If item A4 was fried rice, item B1 was pepper steak, and item C2 was lychee, "meal" would be [fried rice, pepper steak, and lychee]; on the other hand, another perfectly acceptable instance of meal could have been [chicken chow mein, white rice, and sherbert] which would represent a different path through the columns. A similar logic may be applied to defining SKAAs. For example, another way to define perceptual speed could be:

	A	B	C
	Sees	In Less Than	Visual Acuity Test
	Hears	Before	Auditory Reaction Time Test
"perceptual speed" =	Spots	In Under	
	Observes		
	Notices		

where one acceptable instance of perceptual speed is [spots in less than].

If, as is the case in task analysis, the process is actually the reverse and one is presented with an instance which may represent perceptual speed embedded in other information such as:

"I must hear the change in charging pump frequency in less than 10 seconds"

then perceptual speed would be detected in the sentence by using a pattern recognition technique to spot the underlined word combinations and recognize that they correspond to an acceptable path through columns A and B. To go further, however, once perceptual speed is identified, the ability name in turn can function as a pattern that points to an acceptable performance test such as the "auditory reaction time test" in column C.

To accomplish the pattern matches it was helpful to develop rule processing procedures in a computer language suited to manipulation of sentence strings. Because of its ease of use and highly readable code, a simplified version of the LISP language called LOGO was used to initially code the procedures.

SKAA Taxonomy Development. Because readily usable SKAA lists did not exist in Chinese menu definition forms, they had to be generated. Existing taxonomies were surveyed and evaluated as to their usability. It soon became apparent that evaluative criteria for inclusion or exclusion had to be developed and in some cases (for example, cognitive skills) entirely new elements needed to be produced. To facilitate this process a model of skilled human performance was generated. The transformation of existing taxonomic elements into menu forms was accomplished in two steps. First, all available definitions were compiled for taxonomic items along with an analysis of key word patterns which occurred in the examples presented as definition instances. Second, key word patterns were subjected to a computerized thesaurus to find as many equivalent terms as possible. The resulting lists were then screened for applicability and entered into a structured data base. The result was a large set of menu definitions. Since the primary focus of TAPS was to quickly identify tests associated with entry level requirements of NPP operators, lists of usable measurement tests were generated for each ability and rank ordered by factor loadings. TAPS code was written so as to automatically reference these lists whenever a task analysis identified a particular ability as present. In order to illustrate the full potential of the technique, other types of lists were also generated for SKAAs. These lists allow the automated printing of applications, principles, potential safety risks, and even generate customized advice which could be used by an NRC training evaluator or industry training developer. Lists were developed for every taxonomic item; however, a rigorous compendium of human factors information was not attempted within the scope of this effort.

Method Features

Figure 1 illustrates some of an actual output for a sample sentence that illustrates TAPS capabilities. At the top of the figure is the original sentence which shows errors in capitalization, punctuation, and includes technical abbreviations. The second sentence is the result of the first analysis step in which TAPS cleans up obvious errors and expands the abbreviations to their full length. Thus, HPCI becomes high pressure coolant injection, capitalization is normalized, and punctuation is removed. Although the example uses a single sentence, TAPS is not text limited, and works just as efficiently on paragraphs or even multiple pages of typed descriptions.

TAPS systematically outputs skills and the information associated with them, knowledge, then attitudes, and finally abilities. The skill detected in Fig. 1 illustrates the ability of the program to serve as an automated source of guidance to a training developer by listing human factors insights associated with skill categories. "Knowledge" illustrates another capability of the program. After a general knowledge category such as "regulatory guides" was detected, the program retains specific information about the particular instance of regulatory guidance that was found. It then inserts the information into a sentence frame so as to produce customized textual material specific to the task being analyzed. The advice can be as detailed or general as desired, but only very simple principles are used in the present TAPS version. The detection of attitudes illustrates the capability of TAPS to use indirect clues. Since attitudes generally have to be inferred, TAPS recognized that the HPCI was a safety-related system and that the sentence was referring to maintenance behavior. Consequently, an individual's attitude toward "personal responsibility" could have a significant safety impact if maintenance was done unsupervised or in a slipshod fashion. Finally, the "deductive reasoning" ability illustrates that TAPS could be used to produce customized tests in real time.

Conclusions

As the result of efforts by both industry and NRC to assess and improve operator training and qualifications, the U. S. nuclear industry is moving to adaptation of SAT, and the NRC is in parallel moving to adapt the SAT approach in their evaluation of training and qualifications. The research summarized in this paper is intended to provide methods and a technical basis for NRC's evaluation. A framework for an SAT-based evaluation process has been developed and current efforts are underway to develop the specific methods and tools to implement the process.

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EVALUATION OF THE USE OF EXPERT JUDGMENT TO ESTIMATE HUMAN ERROR PROBABILITIES

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ABSTRACT

As part of a broad research program to develop human reliability data and estimates to support probabilistic risk assessment (PRA), the Nuclear Regulatory Commission has supported a program to develop, test, and evaluate the use of expert judgment to estimate human error probabilities (HEPs). During FY 84, this program conducted an empirical test of previously developed procedures based on psychological scaling. This test evaluated the practicality, acceptability, and usefulness of two specific procedures: paired comparisons and direct numerical estimation. A complete report of this research is provided in NUREG/CR-3688. This paper briefly summarizes the two procedures and how they would be implemented. It then describes the empirical test and its results. The results of this test provided a positive evaluation of the use of expert judgment. Judgments were shown to be consistent and to provide HEP estimates with a good degree of convergent validity. Of the two techniques tested, direct numerical estimation appears to be preferable in terms of ease of application and quality of the results. The fact remains, however, that actual relative frequencies of errors are not available, so predictive validity against such a criterion has not been established. In the absence of such data, and given the practical advantages such as the time and cost of using expert judgment, this approach appears to be a feasible approach to obtaining needed HEP estimates for PRAs or other uses.

SECTION 1. INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) with the cooperation and assistance of the nuclear industry has undertaken an effort to improve quantitative estimates of the risks associated with nuclear power plants and to use this information to increase safety. As was demonstrated by Three Mile Island and has been further shown in Probabilistic Risk Assessments (PRAs) for nuclear power plants, human reliability can play a significant role in risk. Therefore, quantitative human reliability estimates are needed.

1.1 Background

To meet this need for quantitative estimates, the NRC is supporting research in several areas potentially able to provide the necessary data:

1. Use of licensee event reports,
2. Computer modeling of human performance,
3. Nuclear power plant simulator experiments,
4. Use of expert judgment.

This paper describes recently completed research on the use of expert judgment to estimate human error probabilities (HEPs) sponsored by the NRC and administered through a contract with Sandia National Laboratories.

Expert judgment has several potential strengths for providing needed estimates. Such estimates can be obtained relatively easily and quickly, with a relatively low cost. The types of tasks for which error probabilities are estimated can be defined in advance based on the needs of the PRA or other requirements. On the other hand, expert judgment may be susceptible to biases and there has been no demonstration of its validity in this particular context.

The NRC research program, administered and monitored by Sandia National Laboratories (SNL), was undertaken to explore the extent to which these potential strengths could actually be realized and to determine to the extent possible, the validity of resulting estimates. An extensive literature review, reported in NUREG/CR-2255 (Stillwell, Seaver, and Schwartz, 1982), was undertaken to determine the potential for the use of expert judgment to estimate human error probabilities based on previous research and applications of expert opinion in other contexts. Then, a detailed set of procedures for five different expert judgment techniques was developed and presented in NUREG/CR-2743 (Seaver and Stillwell, 1983). Following these efforts, SNL contracted with General Physics Corporation and its subcontractor, The MAXIMA Corporation to conduct an empirical evaluation of these procedures. This effort, completed in June 1984, culminated in a two volume report, NUREG/CR-3688 (Comer, Seaver, Stillwell, and Gaddy, 1984), that provides an overview of the procedures tested and the results of the evaluation; as well as appendices giving step-by-step instructions for use of the procedures, a detailed description of the evaluation and statistical analyses, and the actual human error probability estimates that were obtained. This paper summarizes this research and its results.

1.2 Purpose of Research

For this empirical evaluation, two of the five procedures were selected for testing. (A third procedure is being evaluated under a different NRC contract with Brookhaven National Laboratory.) These two procedures, paired comparisons and direct numerical estimation, were selected because they represent the extremes of the expert judgment procedures with respect to the number of experts that are required and the difficulty of the judgments.

In order to conduct a comprehensive evaluation, two sets of issues were developed. The "Program Issues", shown in Table 1, relate to the potential role of expert judgment in estimating HEPs. The "Technical Issues" in Table 2 address how expert judgment procedures should be implemented, if they are shown to adequately satisfy the Program Issues. This research, then, attempted to address these issues and as a result provide the needed evaluation of expert judgment as a means of obtaining quantitative human reliability estimates.

Table 1 Program issues

P1. Do psychological scaling techniques produce consistent judgments from which to estimate HEPs?

P2. Do psychological scaling techniques produce valid HEP estimates?

P3. Can the data collected using psychological scaling techniques be generalized?

P4. Are the HEP estimates that are generated from psychological scaling techniques suitable for use in PRAs and for entry into the Human Reliability Data Bank as described in NUREG/CR-2744 Volume 2 (Comer et al, 1983)?

P5. Can psychological scaling procedures be used by persons who are not expert in psychological scaling to generate HEP estimates?

P6. Do the experts used in the psychological scaling process have confidence in their ability to make the judgments?

Table 2 Technical issues

T1. Based on measures of consistency and comparisons with other human reliability estimates, is there any difference in the quality of estimates obtained from the two techniques?

T2. Is there any difference in the results based on the type of task that is being judged?

T3. Do education and experience have any effect on the experts' judgments?

T4. Based on the number of probability estimates and the functional relationship between the paired comparison scale and the probability scale, how should the paired comparison scale be calibrated into a probability scale?

T5. Can reasonable uncertainty bounds be estimated judgmentally?

SECTION 2. EXPERT JUDGMENT TECHNOLOGY

We are, of course, interested in the systematic use of expert judgment, not in ad hoc intuitive judgments. Specifically, we used psychological scaling which is the process of assigning numbers (in this case HEP estimates) to objects, events, or their properties (here usually operator tasks) in such a way that the numbers represent relationships among them (i.e. the likelihood of human error). Here we briefly describe the two techniques used in the evaluation study, paired comparisons and direct numerical estimation. More detail on these techniques plus the other techniques not empirically tested can be found in NUREG/CR-2743 and NUREG/CR-3688, Appendix A, Volume 2.

2.1 Paired Comparison Technique

Paired comparison scaling is based on judgments such as "a human error is more likely on task a than on task b." Specific, numerical judgments are therefore not required. Experts usually find such judgments relatively easy to make. By obtaining paired comparison judgments between all pairs of a set of tasks from a number of experts, and following the steps outlined in Figure 1, HEP estimates can be obtained and applied.

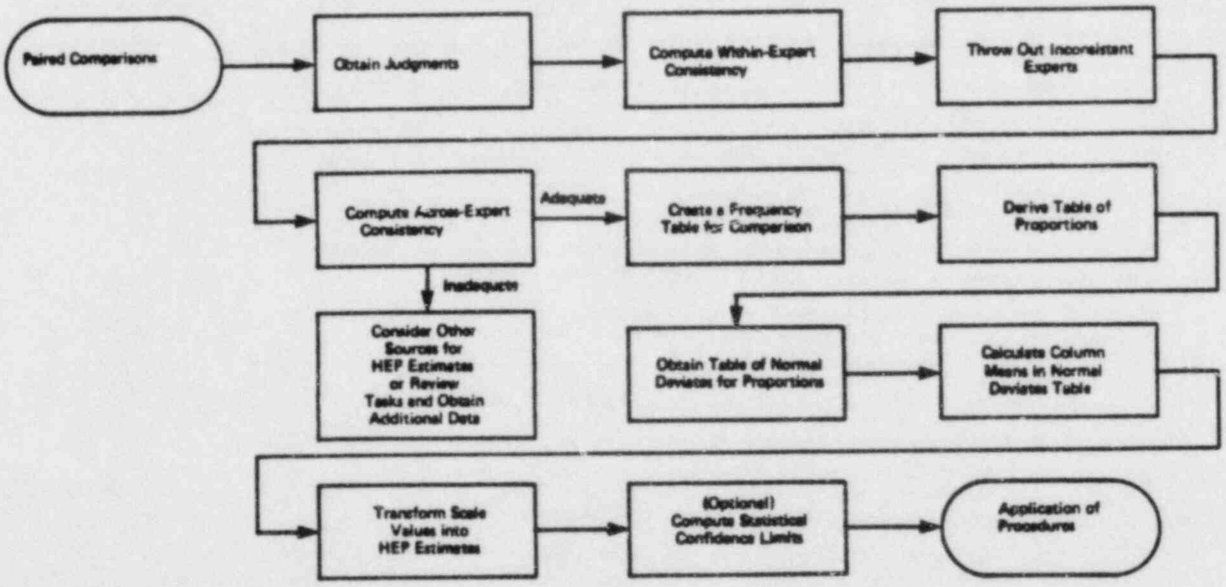


Figure 1 Major steps in using paired comparisons.

This technique first uses traditional paired comparison scaling based on a specific model (cf. Torgerson, 1958) to obtain a numerical subjective scale value for the likelihood of human error on each task. These scale values are not on a probability scale so they must be transformed, or "calibrated," into probabilities. This transformation is accomplished using at least two "anchor tasks," for which HEP estimates are available from another source. The transformation is

$$\log HEP = as + b$$

where log HEP is the logarithm of the desired HEP estimate, s is the scale value, and a and b are constants that are obtained from the anchor task HEP estimates. Details of the procedure for using paired comparisons to obtain HEP estimates can be found in Appendix A of NUREG/CR-3688, Volume 2.

2.2 Direct Numerical Estimation

Direct numerical estimation requires the experts to provide probability estimates for the HEP for each task. HEP estimates are then derived as shown in Figure 2, with the key step being the aggregation of the estimates of individual experts into a single overall estimate for each task.

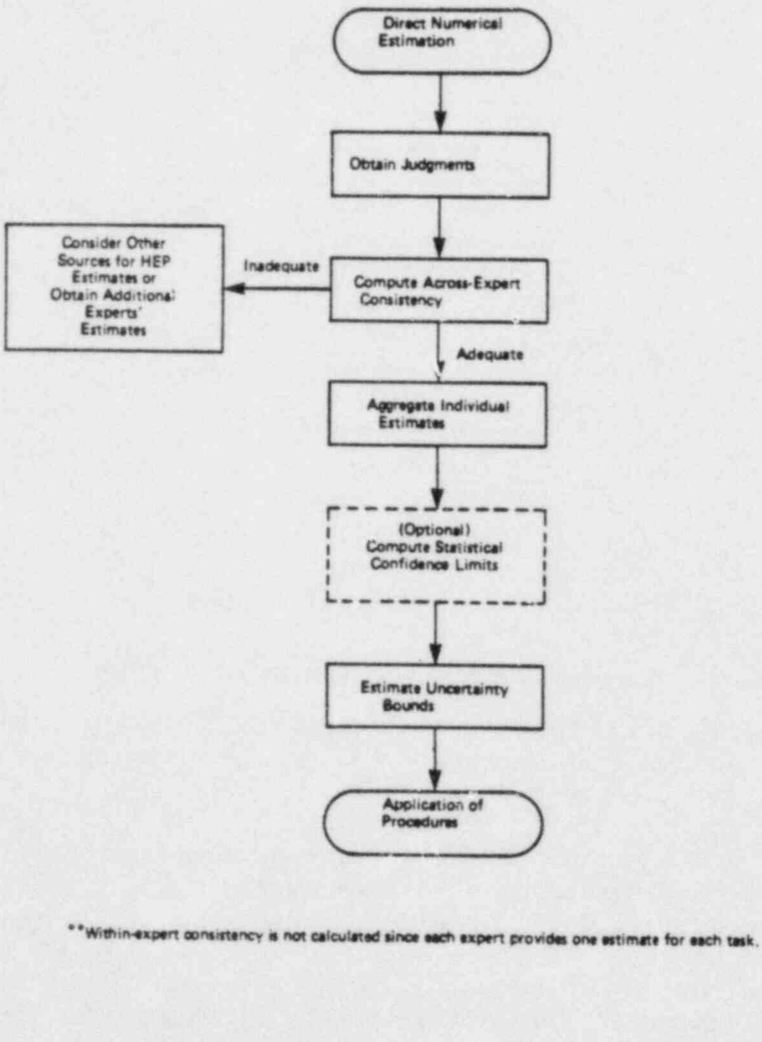


Figure 2 Major steps in using direct numerical estimation.

Another feature of direct numerical estimation is that it can be used to obtain estimates of uncertainty bounds that reflect the probable range over which HEP estimates would vary as conditions such as operator training, plant design, written procedures, etc. change. This feature may be particularly important in situations where worst-case rather than best-estimate risks are being identified. It can also help to identify how much risk reduction could be obtained by improving conditions.

2.3 Implementation

Figure 3 depicts the primary considerations in implementing a procedure based on expert judgment. Step-by-step implementation requirements are given in Appendix A of NUREG/CR-3688. The personnel required are:

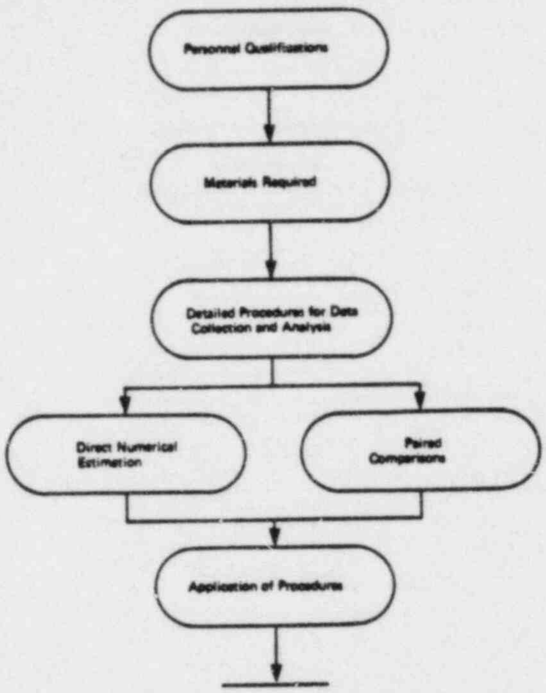


Figure 3 Overview of psychological scaling.

- a human reliability analyst who is familiar with the tasks to be considered and who can translate task definitions from PRA language into terms familiar to the experts;

- at least six subject matter experts (e.g. certified operations instructors) for direct numerical estimation or ten for paired comparisons who are familiar with the tasks to be judged and the range of operator responses to the tasks;

- a data collection session administrator familiar with instructions to be given to the experts (available in Appendix A of NUREG/CR-3688, Volume 2);

- a data analyst with some familiarity with mathematics and statistics.

The roles of human reliability analyst, data collection session administrator, and data analyst can be combined in one or two persons if they are appropriately qualified.

Materials required include:

- well-defined tasks including any assumptions regarding the conditions or performance shaping factors (PSFs) that are to be assumed (e.g. number of operators present, operator's experience, environmental conditions in plant);

- response booklets to make the experts' judgments as easy as possible;

- complete instructions for both the data collection session administrator and the experts;

- materials for data analysis including coding sheets, a calculator with logarithms, and a standard statistics textbook.

With the exception of task definitions, these materials are relatively easy to develop or obtain. Examples are contained in Appendix A of NUREG/CR-3688, Volume 2. Defining the tasks is a very important part of the procedure and should be given careful attention by the human reliability analyst with assistance if possible from PRA personnel and subject matter experts such as operators or operator instructors.

A detailed description of the data collection and analysis procedures would be too extensive for this paper. Such a description, based on the steps in the paired comparison technique and direct numerical estimation shown previously in Figures 1 and 2 respectively, is provided in Appendix A of NUREG/CR-3688, Volume 2. If the judgments collected meet the consistency requirements given there, the resulting HEP estimates are then ready for use in a PRA or for other purposes.

SECTION 3. EVALUATION OF EXPERT JUDGMENT TECHNOLOGY

The primary purpose of this NRC/SNL sponsored effort was to perform an empirical test and evaluation of the expert judgment procedures. Through addressing the specific issues shown previously in Tables 1 and 2, this research was to determine the practicality, acceptability, and usefulness of using expert judgment to estimate HEPs. For the technology to be worth pursuing, relatively positive answers were needed to the following questions:

- Is expert judgment practical to implement in terms of cost and procedural issues?
- Will industry accept the techniques as a viable means of acquiring estimates?
- Will government and industry use expert judgment as part of the PRA process?

3.1 Evaluation Procedure

Two sets of tasks were developed corresponding to Level 1 and Levels 2 and 3 in the Human Reliability Data Bank (Comer, Kozinsky, Eckel, and Miller; 1983). (Research on the Data Bank is described in another paper in this session.) The Data Bank is being developed to organize and structure human actions for which reliability data may be needed for PRAs. Level 1 represents relatively high level actions that combine systems with operator duties. Level 2 combines system components with operator tasks. Level 3 includes controls and displays combined with task elements. For this research, all defined tasks were for BWR plants. Level 2 and 3 tasks were developed so resulting HEP estimates could be compared with estimates from simulator experiments (Beare, Dorris, Bovell, Crowe, and Kozinsky; 1984) and from NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications (Swain and Guttmann, 1983). Fifteen Level 1 tasks and twenty Level 2 and 3 tasks were developed. These tasks are listed in Volume 2 of NUREG/CR-3688.

Nineteen NRC-certified BWR instructors served as the subject matter experts. Instructors were used because of their experience observing a large number of operators including their performance in simulated accident scenarios. These experts had an average of 12.3 years of experience as instructors and operators.

Each expert made both paired comparison judgments and direct numerical estimates (including uncertainty bounds). They also provided some background information, and assessments regarding the ease of the judgments required and their accuracy.

3.2 Evaluation Results

This study provided a positive evaluation of both expert judgment techniques used to estimate HEPs. Although all aspects of the issues could not be completely resolved on the basis of this single test, it has provided substantial support for the use of expert judgment.

Table 3 shows the estimated HEPs for the Level 1 tasks. The paired comparison technique used both two and four direct estimates for the "anchor tasks" to transform scale values into probabilities. Table 4 shows the estimates for Level 2 and 3 tasks. For these estimates, in addition to varying the number of anchor tasks used, the source of HEP estimates for the anchor tasks was varied including direct estimates, Handbook (NUREG/CR-1278), and simulator estimates. Perusal of these tables indicates relatively good agreement among the different estimates. The practical significance of these results indicated by these tables and also by figures 5 through 8, is that, for the most part, the differences in estimates are within one order of magnitude of each other.

Task	Direct Numerical Estimation	Paired Comparisons	
		2 Anchors	4 Anchors
1	0.0007	0.0003	0.0003
2	0.001	0.0006	0.0007
3	0.0008	0.0004	0.0004
4	0.0002	0.0002	0.0002
5	0.0002	0.0002	0.0003
6	0.07	0.07	0.06
7	0.006	0.003	0.003
8	0.04	0.05	0.04
9	0.0001	0.0001	0.0002
10	0.01	0.001	0.001
11	0.0003	0.0002	0.0003
12	0.001	0.002	0.002
13	0.002	0.001	0.001
14	0.0005	0.0004	0.0005
15	0.03	0.04	0.03

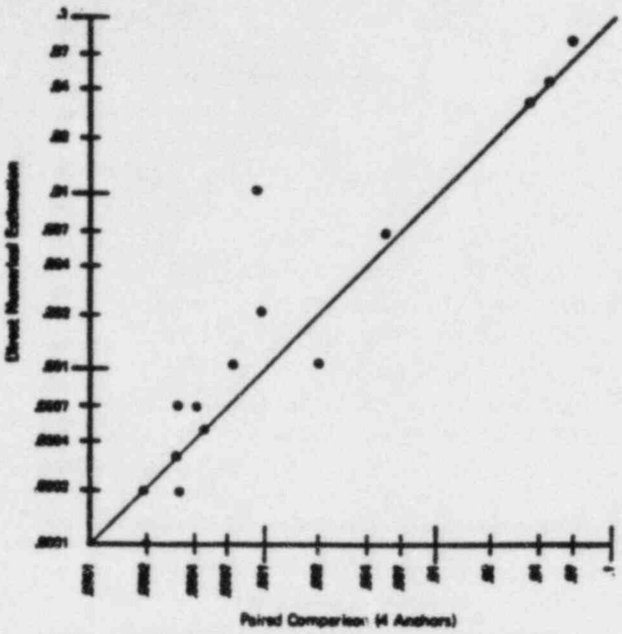
Table 3 Comparison of HEP estimates for Level 1 tasks.

Figure 5 plots directs numerical estimates versus paired comparison estimates with four anchors for Level 1 tasks. (The results with two anchors are very similar.) Figures 6 through 8 show selected similar plots for Level 2 and 3 tasks. The correlations for these comparisons are .94 for Level 1 tasks (direct estimates versus paired comparisons); and for Level 2 and 3 tasks, .89 for direct estimates versus paired comparisons, .68 for direct estimates versus Handbook estimates, and .40 for paired compairsons estimates versus Handbook estimates. (All correlations are significant at the .001

level except the .40 which is significant at the .05 level.) Generally, the agreement is good, particularly between direct estimates and paired comparison estimates. Other plots are similar, except for paired comparisons estimates with simulator anchors which tended to cluster together because of the limited range of the HEP estimates for simulator anchor tasks. (Only four estimates were available so there was no choice of anchor tasks.)

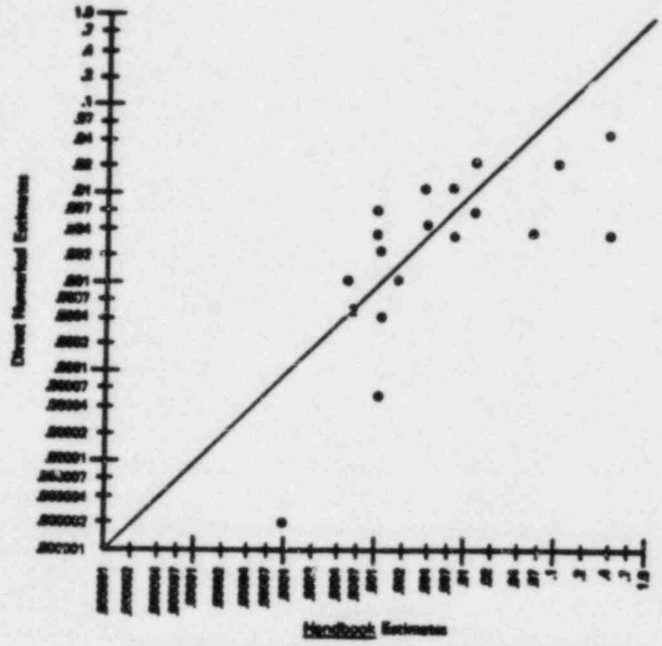
	HEP Estimates								
		Paired Comparisons							
		Direct Numerical Estimation		Handbook		Simulator			
Task	Direct Numerical Estimation	2 Anchors	4 Anchors	2 Anchors	4 Anchors	2 Anchors	4 Anchors	Handbook	Simulator
1	0.004	0.003	0.006	0.03	0.007	0.007	0.003	0.003	0.007
2	0.002	0.001	0.002	0.01	0.004	0.004	0.003	0.001	0.0006
3	0.0005	0.00002	0.00005	0.0006	0.0002	0.0005	0.001	0.0005	0.0005
4	0.0005	0.00005	0.0001	0.001	0.0005	0.0009	0.001	0.0005	0.006
5	0.02	0.02	0.02	0.12	0.02	0.02	0.004	0.01	
6	0.0004	0.000007	0.00002	0.0003	0.0001	0.0003	0.001	0.001	
7	0.001	0.00006	0.0002	0.002	0.0006	0.001	0.002	0.0005	
8	0.006	0.006	0.01	0.05	0.01	0.01	0.004	0.001	
9	0.01	0.01	0.02	0.11	0.02	0.02	0.004	0.003	
10	0.003	0.003	0.005	0.03	0.007	0.007	0.003	0.005	
11	0.003	0.0001	0.0003	0.003	0.0009	0.001	0.002	0.25	
12	0.02	0.007	0.01	0.06	0.01	0.01	0.004	0.10	
13	0.007	0.003	0.005	0.03	0.006	0.006	0.003	0.01	
14	0.000002	0.000002	0.000006	0.0001	0.00006	0.0003	0.0008	0.0001	
15	0.04	0.04	0.06	0.25	0.04	0.03	0.005	0.25	
16	0.00005	0.00001	0.00004	0.0005	0.0002	0.0005	0.001	0.001	
17	0.001	0.0001	0.0003	0.003	0.0008	0.001	0.002	0.002	
18	0.01	0.03	0.04	0.20	0.03	0.02	0.005	0.006	
19	0.003	0.0008	0.002	0.01	0.003	0.004	0.002	0.05	
20	0.003	0.001	0.002	0.02	0.004	0.005	0.003	0.001	

Table 4 Comparison of HEP estimates for Level 2 and 3 tasks.



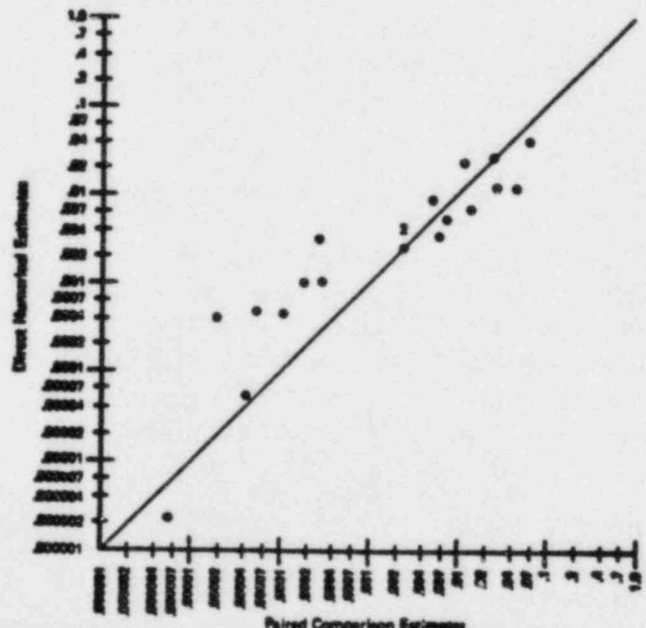
Level 1 direct numerical estimates and paired comparison estimates with four anchors.

Figure 5



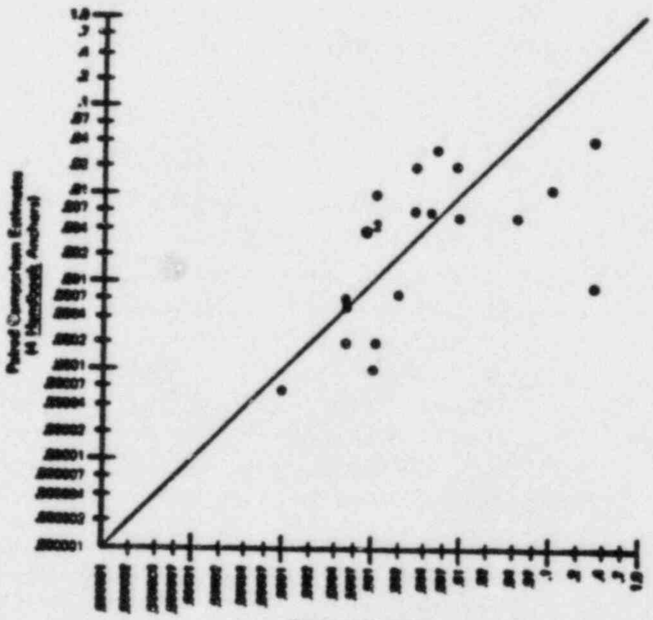
Level 2/3 direct numerical estimates and Handbook estimates.

Figure 6



Level 2/3 direct numerical estimates and paired comparison estimates with four direct estimate anchors.

Figure 7



Level 2/3 paired comparison estimates with four Handbook anchors and Handbook estimates.

Figure 8

2 = two points coincide

Additional results are summarized below as they relate specifically to the Program and Technical Issues addressed.

1. Do Psychologocial Scaling Techniques Produce Consistent Judgments From Which to Estimate HEPs?

Two types of consistency were investigated: the internal consistency of each expert's paired comparison judgments, and the across-expert consistency for both paired comparison and direct estimate judgments. Internal consistency was measured using the coefficient of consistency which is based on the number of intransitive triads (a is more likely than b, b more likely than c, and c more likely than a) relative to the total possible number of intransitive triads. A value of zero indicates the maximum inconsistency and a value of one represents complete consistency. For both sets of tasks internal consistency was extremely high ranging from .67 up to .99 with almost all coefficients above .8. Mean coefficients were .89 for Level 1 tasks and .86 for Level 2 and 3 tasks.

Across expert consistency was also good as measured by the coefficient of concordance which ranges from zero--no agreement--to one--complete agreement among the experts. For paired comparisons, these coefficients were .54 and .57 for Level 1 and Level 2 and 3 tasks respectively. For direct estimates, the corresponding coefficients were .39 and .42. Coefficients for the estimated uncertainty bounds were somewhat lower ranging from .34 to .41. All coefficients were highly significant.

2. Do Psychological Scaling Techniques Produce Valid HEP Estimates?

Analyses focused on convergent reliability--the extent to which different approaches to estimating HEPs produce the same estimates--since there were no available relative frequency HEP estimates against which predictive validity could be tested. In addition to the correlations described above, several additional analyses based on analysis of variance (ANOVA) substantiate the agreement among different approaches. The primary differences that did occur were in paired comparison estimates derived using different anchor tasks when only two anchors were used. Using four anchors greatly reduced the effects of the anchors on HEP estimates.

3. Can Data Collected Using Psychological Scaling Techniques Be Generalized?

Generalizability was determined primarily by the specification of tasks. Level 1 tasks were selected and defined to be generic for all BWR plants, so HEP estimates should be appropriate with adjustments (e.g. within the uncertainty bounds) for plant-specific factors. Level 2 and 3 tasks should be somewhat more generalizable (i.e. also appropriate for BWR plants) because they are not system-specific and are found in most plants.

4. Are HEP Estimates That Are Generated From Psychological Scaling Techniques Suitable For Use in PRAs and the Human Reliability Data Bank?

This issue was also addressed primarily in the definition of tasks, and also by the fact that judgments obtained were consistent and satisfied convergent validity requirements. Since the Data Bank was designed to be consistent with PRA needs, use of tasks from multiple levels of the Data Bank confirms the suitability of the generated HEP estimates. An additional important consideration was the capability to produce estimates of uncertainty bounds. Although because of space limitations the results with respect to estimation of uncertainty bounds cannot be discussed fully here (see Appendix B in NUREG/CR-3688, Volume 2), the analyses that were conducted indicated reasonably good expert consistency and convergent validity in the direct estimates of uncertainty bounds.

5. Can Psychological Scaling Procedures Be Used By Persons Who Are Not Expert in Psychological Scaling to Generate HEP Estimates?

All procedures have been designed and adequately documented so that an expert in psychological scaling need not be available for them to be used. These procedures, including instructions were pretested, revised, and then used in actual data collection. No psychological scaling expert participated actively in data collection, and no difficulties were encountered. The expertise that is required was described above in section 2.3.

6. Do the Experts Used in the Psychological Scaling Process Have Confidence in Their Ability to Make the Judgments?

The confidence of experts in their judgments is one indication of the reasonableness of the estimates, although experience in other contexts suggests that experts can make good probability estimates even though they doubt their own ability to do so. When questioned regarding the accuracy and difficulty of their judgments, the experts were generally neutral, considering their judgments neither particularly accurate nor inaccurate, and neither easy nor difficult. Paired comparisons judgments were considered somewhat more accurate than direct estimates. Six point scales were used to obtain the expert's opinions with 1 indicating accurate or easy judgments and 6 indicating inaccurate or difficult judgments. For accuracy, mean ratings were 2.1 and 3.1 for paired comparisons and direct estimates respectively (3.5 is neutral). Mean difficulty ratings were 3.2 and 3.3 respectively. Uncertainty bounds estimates were considered somewhat less accurate (mean 3.5) and more difficult (mean 3.9).

7. Is There Any Difference in the Quality of Estimates Obtained From the Two Scaling Techniques?

There was relatively little difference in the HEP estimates obtained from the two techniques. Paired comparisons had a somewhat higher across-expert consistency, while direct estimates correlated somewhat higher with Handbook estimates. Experts did perceive their paired comparison judgments to be somewhat more accurate. None of these differences provide a strong basis for selecting one technique over the other. Therefore, as is discussed below, selection of a specific technique to be used can be based on practical considerations such as the number of experts required and the need for estimates of uncertainty bounds.

8. Is There Any Difference in the Results Based on the Type of Task That is Being Judged?

Study results were generally similar for the two task sets. Internal consistency and convergence were slightly higher for Level 1 tasks while across-expert consistency was somewhat higher for Level 2 and 3 tasks. Again, the differences are not large. An important finding for both task sets was that the experts considered the tasks relatively easy to understand (mean rating 1.7 for both sets on a six point scale with 1 indicating easy to understand).

9. Do Education and Experience Have Any Effect on the Experts' Judgments?

The experts used in this study were quite homogeneous with respect to their eduction, amount of experience, and type of license or certification. As a result, these variables were not related in any way to the judgments. Whether similar results would be obtained from more heterogeneous experts is an open question.

10. How Should the Paired Comparison Scale Be Calibrated Into a Probability Scale?

Our results indicated that the logarithmic relationship described in section 2.1 was more appropriate than a linear relationship for transforming scale values into probabilities. In addition, four anchor tasks rather than two provided better convergence among estimates.

11. Can Reasonable Uncertainty Bounds Be Estimated Judgmentally?

The experts were able to estimate uncertainty bounds using direct estimation, but these estimates could be subjected to only limited analyses because of limitations in the study design. Thus, while the results were somewhat positive, they were insufficient to provide a definitive answer to this question.

4. CONCLUSIONS AND POTENTIAL USES

The conclusions drawn from this evaluation effort were:

- Both direct numerical estimates and paired comparison judgments more than met statistical requirements for consistency.

- Convergent validity of the HEP estimates was good, particularly if the effects of using only two anchor tasks for paired comparison estimates are disregarded. It should be noted though, that predictive validity with respect to HEP estimates based on the actual relative frequency of errors could not be established because of the lack of such estimates. (This will be a difficulty in validating any procedure used to estimate HEPs.)

- The tasks and their HEP estimates should be generalizable to all BWRs. Results should also be somewhat generalizable to other, similar groups of experts. The actual extent of this latter generalizability has not been fully tested.

- Tasks can be appropriately defined and HEP estimates for them can be obtained so that the estimates can be used in PRA and in the Human Reliability Data Bank.

- The judgments required can be obtained from experts without the use of an expert in psychological scaling. However, expertise in human reliability, statistics, and task subject matter is needed for task selection, analysis, and judgment.

- Experts making the judgments have only a moderate degree of confidence in their judgments. (Often experts without experience in making these types of judgments will lack confidence in the judgments. Confidence will increase with experience. Lack of confidence does not imply that the judgments are not sound.)

- Only minor differences occur in the evaluations of direct numerical estimates and paired comparison estimates. One technique cannot be selected over the other on the basis of these analyses alone. In some situations, use of direct numerical estimation may be preferred to paired comparison scaling because of practical considerations such as requiring fewer experts (as few as six for direct estimation versus 10 to 12 for paired comparison) and less of the experts' time. For example, if paired comparison scaling is used to obtain uncertainty bound estimates, it will increase the amount of time required to make judgments by three (once for the HEP estimate, once for the lower bound, and once for the upper bound).

- Only minor differences in consistency and convergent validity occurred in the results for the two types of tasks (Level 1 and Levels 2 and 3). Expert judgment can be used to estimate HEPs for either type of task.

- Background variables such as education, experience, and type of license/certification did not affect judgments. The extent to which this conclusion is true, beyond the specific group of instructors used as experts in this study, is not known because the group used was very homogeneous.

- For paired comparison estimates, scale values should be transformed into HEP estimates using a logarithmic relationship. Human error probability estimates for more than two tasks (e.g., four) should be used to estimate the parameters in the transformation.

- Uncertainty bounds can be estimated using direct estimates, although this study was not designed to thoroughly test the resulting estimates.

As a practical matter, this study demonstrated that either technique can be used to estimate HEPs in a timely manner. Expert judgment data can be obtained and used in a relatively cost-effective manner with tasks that are carefully defined to meet PRA needs. Psychological scaling techniques can thus be used to generate data without some of the difficulties of task definition or inadequate data that may affect simulator studies or field reporting. The main drawback presently in the use of expert judgment or any other procedure to estimate HEPs is the inability to establish predictive validity.

Taken together, the conclusions indicate that these techniques using expert judgment should be given strong consideration for use in developing estimates for the Human Reliability Data Bank. In addition, they can be implemented, as needed, to provide HEP estimates for PRAs. Another potential use of these procedures would be design studies where human error is a consideration. For example, in human factors control room design reviews, human factors problems could be ranked in terms of their potential contribution to human error, which would assist in prioritizing corrective actions.

Additional research on the use of expert judgment might be especially valuable in several areas: time-response functions, estimation of uncertainty bounds, assessment of predictive validity, and development of anchor task estimates. Time-response functions show the probability that an operator will successfully perform a task within a certain time frame, with the probability varying as the amount of time varies. The HEP estimates obtained in this study were essentially estimates for a single point in time. Time response functions provide the data needed for a wider range of contexts. If expert judgment can be used to obtain time-response functions, the number of overall judgments required could be reduced.

In this project, uncertainty bounds estimates were obtained using expert judgment, although this study was not designed to thoroughly test the resulting estimates of bounds. Additional research could be undertaken to explore whether there are systematic biases in these estimates and to further investigate other judgmental methods for obtaining estimates of uncertainty bounds. Finally, simulator studies or other more cost-intensive research could provide a source of a few anchor task HEP estimates needed for paired comparison estimates.

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THE SLIM-MAUD RESEARCH PROGRAM: DEVELOPMENT OF A MULTI-ATTRIBUTE UTILITY BASED METHODOLOGY FOR HUMAN RELIABILITY EVALUATION*

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Introduction

The purpose of this paper is to describe the research program devoted to the development of SLIM-MAUD, a multi-attribute utility-based methodology for estimating human reliability in nuclear power plant. The SLIM-MAUD research program is one of two multiyear programs sponsored by the NRC to determine the applicability of expert judgment techniques to estimating human reliability. The SLIM-MAUD program was implemented through the combined efforts of Brookhaven National Laboratory, Human Reliability Associates, Inc. of Lancashire, England and The London School of Economics and Political Science.

Background

Basic SLIM

The basic rationale underlying SLIM (Success Likelihood Index Methodology) is that the likelihood of an error occuring in a particular situation depends on the combined effects of a relatively small set of performance shaping factors (PSFs). In brief, PSFs include both human traits and conditions of the work setting that are likely to influence an individual's performance. Examples of human traits that "shape" performance might include the competence of an operator (as determined by training and experience), his/her morale and

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motivation etc. Conditions of the work setting which might affect performance include the time available to complete a task, task performance aids, etc. It is assumed that an expert judge (or judges) is able to assess the relative importance (or weight) of each PSF with regard to its effect on reliability for the task being evaluated. It is also assumed that, independent of the assessment of relative importance, the judge(s) can make a numerical rating of how good or how bad the PSFs are in the task under consideration, where "good" or "bad" mean that the PSFs will either enhance or degrade reliability.

Having obtained the relative importance weights and ratings, these are multiplied together for each PSF and the resulting products are then summed to give the Success Likelihood Index (SLI). The SLI is a quantity which represents the overall belief of the judge(s) regarding the positive or negative effects of the PSFs on the likelihood of success for the task under consideration. By assuming that because of their knowledge and experience the judge(s) have a correct idea of the effects of the PSFs on the likelihood of success, then the SLI can be expected to be related to the probability of success or failure that would be observed in the long run in the situation of interest (i.e., the actuarially determined probability). The calculation formula for the SLI is as follows:

$$SLI_j = \Sigma\ W_i \cdot R_{ij}$$

where SLI_j = SLI for the jth task

W_i = normalized importance weight for the ith PSF ($\sum W_i = 1$)

R_{ij} = scale value (rating) of the jth task on the ith PSF.

A major assumption of the SLIM approach is that a SLI generated by this process bears a consistent relationship to the expected long-term probability of failure and can be converted to it in a simple manner. The relationship is assumed to be of the form: Log (probability of failure) = a SLI + b.

SLIM Linked to MAUD

MAUD (multi-attribute utility decomposition) is a flexible, interactive computer based system which can be used to implement SLIM. MAUD was originally developed by the Decision Analysis Unit of the London School of

Economics and Political Science for use in a wide variety of decision analysis problems.

The linking of MAUD with SLIM represents a more sophisticated way of eliciting from judges the rating and weighting information required in the basic SLIM approach. Furthermore, the elicitation procedures of MAUD are in closer accord with the theoretical assumptions underlying the SLI methodology than is the case for the basic SLIM procedure. The MAUD-based implementation has the additional advantage of being able to deal with the evaluation of up to 10 tasks in the same session. It employs MAUD's built-in checks to monitor any dependencies between PSFs which may be present. The MAUD system is fully interactive and is sufficiently "user friendly" such that it can be used unsupervised by individuals or groups of judges with minimal training in computer-based techniques. An example of the dialogue used in SLIM-MAUD and a detailed technical description of the technique is given in Embrey et al. (1984). An important feature of MAUD is the fact that it allows the judges to constantly modify their assessments as their understanding of the situation develops.

In a typical SLIM-MAUD session, the system first asks the judge(s) to name the various tasks for which HEPs are required. It is assumed that the SLIs for all the tasks being assessed in a particular session can be determined by the same PSFs with the same relative weights. At least two reference tasks for which HEPs are available need to be included in the session for calibration purposes. SLIM-MAUD then interactively elicits the PSFs which are relevant in determining the probability of success. MAUD performs a comprehensive set of consistency checks on the judges' use of these PSFs in assessing the tasks under consideration. This process is repeated with the various combinations of tasks to generate a series of factors which are equivalent to the PSFs that are elicited directly in the basic SLIM technique.

With SLIM-MAUD, judges first rate tasks and then weight them, thus reversing the order used in the basic SLIM elicitation technique. Judge(s) are first asked to rate each of the tasks on nine-point scales and define their "ideal" point on each scale, i.e., the rating scale value which would be optimal in promoting success. MAUD uses this information to re-scale the PSFs so that increasing scale values always indicate increasing likelihood of success. This is necessary because with some PSFs, e.g. stress, high or low values degrade the probability of success, whereas moderate values increase it.

The next step in SLIM-MAUD develops the PSF weights by comparing pairs of tasks which have different values on two of the PSF scales. SLIM-MAUD asks the judges which of the two tasks would be most likely to succeed, and then iteratively "degrades" one of the PSF ratings of the task judged most likely to succeed and improves one of those of the task less likely to succeed. This process is repeated until the judges' opinions reverse themselves with respect to which of the two tasks is most likely to succeed. By repeating this process for a range of PSFs, SLIM-MAUD is able to determine the relative weights of the various PSFs for the task set under consideration, as perceived by the judge(s). From the weights and ratings SLIM-MAUD then calculates the SLIs for each task.

A separate computer program is then used to convert the SLI values into human error probabilities (HEPs) using the calibration equation derived from the two reference tasks.

Advantages of the SLIM Approach

A major advantage of SLIM compared with other judgment-based methodologies is that it explicitly identifies the factors (PSFs) which are judged to be major determinants of the probability of error in the tasks being assessed.

The weights assigned to these PSFs can be used to provide design recommendations by identifying which changes will have the greatest effect in reducing the likelihood of error. It is also possible to conduct sensitivity analyses, where the effects of postulated changes in PSFs on the overall expected likelihood of success can be evaluated. This is done by simply varying the ratings for the PSFs of interest.

Another advantage of the SLIM approach is that it is highly scrutable, i.e. the means via which the final result is arrived at are accessible to external audit and review.

Phases of the SLIM-MAUD Program

The SLIM-MAUD research program was implemented in four phases which are summarized in Table 1.

Table 1. Four-Phase SLIM-MAUD Research Program

Phase	Type of Research	Objective
I	Experiment	To determine if empirical data supports the assumed log relationship in basic SLIM
II	Field Study	To evaluate the practicality of implementing basic SLIM with actual nuclear power tasks
III	Linking of computer software	To assess the feasibility and practicality of the SLIM to MAUD linkage
IV	Test of SLIM-MAUD	To determine the practicality, acceptability and usefulness of the MAUD-Based implementation of SLIM

Phases I, II, and III have been completed and are fully described in Embrey et. al. (1984) Volumes I and II. In brief, the results obtained in each phase were generally successful in meeting the research objectives established. Phase IV, a comprehensive test of the MAUD-based implementation of SLIM, is currently being implemented. SLIM-MAUD will be evaluated on the basis of three criteria--practicality, acceptability, and usefulness. The three criteria and the specific issues they comprise are summarized in Table 2.

Conclusion

The results obtained thus far from the SLIM-MAUD research program indicate that the methodology is a viable approach to estimating human reliability in nuclear power plants. Furthermore, SLIM-MAUD is a useful diagnostic tool for identifying those PSFs having the most impact on human error.

Table 2

Issues	Methods/Data	Analysis
Practicality:		
Cost	Actual costs incurred for implementating Test Plan.	Costs summation plus discussions of potential cost additions or reductions.
Subject Matter Experts	If feasible, by examining three expert groups: PRA specialists, operators or trainers, and engineers.	Multi-dimensional Scaling (MDS) of user responses.
Support Requirements	Ennumeration of equipment and other materials needed to implement Test Plan.	Discussion of equipment used and other equipment capable of using MAUD.
Transportability	Test will likely be implemented in more than one location.	Experience in setting up and running SLIM-MAUD in separate locations.
Expandability	Development of categorization scheme.	Cluster analysis of user responses.
Time Requirements	Actual experience gained in implementing Test Plan.	Discussion of experience, time considerations and factors affecting time.
Interface With Reliability Data Bank	Ensured by tasks to be evaluated.	None needed.
Implementability of Procedure	Use of more than one session facilitator.	Comparison of the degree of difficulty experienced by different facilitators.
Acceptability:		
Scientific Community	Professional journal submission.	Reviewer comments and/or acceptance of articles.
Expert Participants	Debriefing interview and survey.	Evaluation of interviews and analysis of survey data.
Potential Users	Informal survey.	Evaluation of responses.
Nuclear Regulatory Commission (NRC)	None.	None.
Nuclear Utilities	None.	None.
Usefulness:		
Reliability	Inter-judge consistency.	Use of MDS to assess consistency between individual results.
Face Validity	Survey of expert participants, informal survey of potential users.	Evaluation of open-ended comments and analysis of survey data.
Convergent Validity	Comparison with HEP estimates provided by other subjective techniques.	Examination of magnitude of differences.

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5. Conclusions

This project will arrive at two useful goals. First, the database developed by BNL to include all results reported in PRAs will be useful to regulators, system designers, utilities, and PRA analysts in identifying and articulating PRA human factors data. It is designed to be expandable and will accomodate a vast range of human factors data expected to be useful. Second, the process and structure of PRAs will be improved to make these documents more useful than is currently the case. Both of these goals will make future PRAs useful for a broad range of applications which are not currently available and assist NRC in resolving human factors safety issues.

MAintenance Personnel Performance Simulation (MAPPS) Model: Overview and Evaluation Efforts*

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Background

Over the past three years, the U. S. Nuclear Regulatory Commission (NRC) has sponsored a dedicated program focused on the development of a computerized simulation model for assessing maintainer performance reliability within the nuclear power plant (NPP) maintenance context. The primary impetus of this program was the need for and lack of a comprehensive source of human reliability data for input to probabilistic risk assessment (PRA) studies. Because of the relative paucity of research directed toward the empirical evaluation of maintainer performance, the lack of human reliability data in the maintenance context is even more pronounced than for other contexts such as NPP operation. The end-product of this program, an evaluated computer simulation model entitled "MAPPS" (MAintenance Personnel Performance Simulation), has been developed and is currently being evaluated. Initial model runs and preliminary comparisons to gathered evaluation data indicates that MAPPS will provide a practical, acceptable and useful tool for generating valuable maintainer performance data for input to PRA studies. The versatility generally associated with simulation-type models is also inherent in the MAPPS model and allows its usefulness to extend beyond its primary purpose for PRA. As such, the MAPPS model represents a comprehensive framework for analyzing maintenance activities. Not only will the model provide a quantitative source of performance data for PRA purposes and for making decisions related to maintenance activities, it will also provide the user/analyst with qualitative insights into the design and structure of maintenance tasks and the maintenance context in general. Following a brief description of the MAPPS model and a discussion of model evaluation efforts, various potential applications of the model will be presented.

The MAPPS Model

The MAPPS model is of the simulation type and has been specifically developed to simulate NPP maintenance activities. One of the main advantages of simulation modeling is that it can be effectively utilized to model complex systems composed of a large number of interdependent variables. Efforts to model such systems deterministically tend to be extremely difficult due to the complexity of expressing analytically the numerous and highly interdependent relationships between system variables. Monte Carlo techniques used in simulation modeling allow these relationships to be addressed stochastically and aid in retaining the dynamic realism of the system to be modeled.

The development of the MAPPS model was firmly based on information provided by a front-end analysis[1] and four job analyses.[2,3,4,5] The initial model design was subjected to a review by subject-matter experts and a second review was subsequently carried out after model development was completed and a plan for model evaluation was formulated. The modeling efforts and plans for evaluation received the endorsement and positive support of the review panels.

MAPPS is a task-oriented simulation model that includes environmental, motivational, task and organizational variables which influence personnel performance reliability. It yields information such as predicted errors, personnel requirements, areas of maintainer stress and fatigue, performance time and required maintainer ability levels for any corrective or preventive maintenance actions in NPPs.

The MAPPS model was developed to be rich in both input variables and output parameters. The generous assortment of input variables provides the user with a high degree of flexibility in describing the maintenance situation of interest and for performing parametric sensitivity analyses of important maintenance variables. The large spectrum of output parameters provides the user with an abundance of data that allows significant insight into the performance of the task under consideration. User-friendly interactive menus were developed for MAPPS that allows fast and easy control of the input data and choice of relevant output format types. The MAPPS model was also developed to be minimally dependent upon data base information relating to average subtask completion times. Such data are required by MAPPS as input but are often not available in data banks or from observational data. Accordingly, a rank-ordering regression technique was developed for estimating this data for all subtasks from a minimum of six actual duration times entered by the user. Comparison of the output of this method with independent actual maintenance subtask performance time measures indicated very close correspondence between the actual time measures and the time estimates produced by the regression technique.[6]

Although a full in-depth description of the MAPPS model is beyond the scope of the present paper, an overview of the organization and content of MAPPS is presented. Interested readers are referred to a two-volume NUREG/CR report[7,8] which describes in considerable detail the structure and content of MAPPS.

A summary of MAPPS functional processing is presented in Fig. 1. Steps 6 through 11 identify an iteration loop that simulates the entire task of interest the number of times specified by the user. A number of iterations (simulations) of a task is necessary to smooth the random effects introduced by various stochastic processes within a simulation. Various hierarchial levels of output may be generated by MAPPS (subtask, shift, iteration and run) if requested. Output at the run level (overall task level) is the only output that is always generated for the user. Step 7 of Fig. 1 points out that MAPPS will select the required maintainers for each subtask from the maintainer work group identified by the user. When more than one maintainer exists for a given maintainer type, MAPPS will choose the maintainer with the least time worked during the current work shift. Selection of the subtask work crew will also be dependent upon overmanning or undermanning parameters as designated by the user. Step 13 indicates that MAPPS also allows shift changes to occur during task simulation. Shift changes are allowed to occur at specified times during task simulation, or following particular subtasks.

Step 8 of Fig. 1 represents the heart of the MAPPS model. The logic for this step is expanded in Fig. 2 and is utilized for each normal subtask for each iteration of the task of interest. MAPPS addresses 28 subtask kinds, 22 of which are considered normal subtasks and 6 are considered special. Special subtasks such as decision making and troubleshooting are not models of cognitive behavior per se, but are means of determining the time and probability of reaching a correct solution dependent on factors such as intellective ability, goal importance, trouble report quality, etc. The special subtasks will not be discussed further in this paper; rather, processing of the normal subtasks will be emphasized.

The overall purpose of the logic presented in Fig. 2 is to determine: (a) whether or not the subtask is performed successfully, (b) whether or not an undetected error exists in the work, and (c) how much time is involved in completing the subtask. By way of overview, Fig. 2 indicates that the simulation logic is primarily centered around the difference between the current ability levels of the maintainers and the ability requirements of the subtask to be performed. The degree of ability difference has a succinct effect upon the probability of subtask success, the average duration of subtask performance and the total stress experienced by the maintainers. The total stress on the simulated work group is affected by the ability difference value as well as by time stress, radiation stress and communication problems (if present). These other contributors to total stress are described as follows: time stress - when the time required to perform all remaining subtasks is greater than the time available for total task completion, time stress is assumed to be present; radiation stress - when the absorbed radiation dose for a given technician will be greater than 800 millirem (mrem) during the course of task completion, radiation stress is assumed to exist (the NRC's maximum permissible quarterly dose is 1250 mrem); communication stress - if communication is an ingredient in subtask performance, the stress resulting from any communication degradation such as elevated noise level, excessively lengthy communications or large work crews is considered.

The MAPPS model allows the user to specify if an acceptability check is made of the work performed by the simulated maintainers. If so, a supervisor or quality assurance check is simulated in addition to the normal checks made by the work group. Four subtask outcomes are possible: (a) success - the subtask was completed in an acceptable manner without any uncorrected errors; (b) false alarm - the subtask was completed without any uncorrected errors but was found to be unacceptable by a supervisor/QC check; (c) detected error - uncorrected errors existed at the end of subtask performance and were detected by a supervisor/QC check; (d) undetected error - uncorrected errors existed at the end of subtask performance but were not detected by a supervisor/QC check. Perceived work crew failure on a subtask (false alarm, detected error) leads to re-performance of the subtask which in turn leads to higher subtask performance time, higher time and radiation stress and a greater fatigue level. Sequences of subtask failures or successes will also lead to a respective decrease or increase in the maintainers motivation (aspiration) levels. Under the circumstances of high time stress the MAPPS model will allow subtasks of low importance (e.g., clean up subtasks in some instances) to be skipped by the work crew.

The previous several paragraphs have presented a relatively mechanistic view of the functioning of the simulation modules as presented in Fig. 2. Additional qualitative insight into the operation of the model are summarized in the following set of axioms upon which the subtask simulation is based:

1. Subtask success probability and performance duration vary as a function of the difference between the ability requirements of a subtask and the actual ability of the maintainers. As the abilities of the maintainers approach or exceed the ability requirements of a subtask, the subtask success probability increases and the performance time decreases.

2. Stress on the maintainers affects success probability and performance duration. "Moderate" stress increases subtask success probability and decreases performance time. "High" stress (i.e., stress above the stress thresholds of the members of the simulated work group) decreases the success probability.

3. When the workplace temperature exceeds 80° Fahrenheit, performance will degrade as a function of the level of heightened temperature.

4. When maintainers know that the radiation level to which they will be exposed during task performance is such that their total absorbed dose will be greater than their quarterly allowance, they will tend to increase their work pace (to decrease their exposure).

5. Poor component accessibility, inferior procedural aids and protective clothing tend to make maintainer performance slower and less accurate.

6. Fatigue and non-recent performance of a task negatively affect performance time and work quality.

7. The supervisor's requirements relative to work quality will determine whether or not a work group's performance of a subtask is "acceptable" or "unacceptable."

8. Work groups with high levels of aspiration working for supervisors with high levels of aspiration will perform more quickly and thoroughly.

9. A favorable organizational climate reinforces productivity.

10. If communication is required during the course of the performance of a subtask, subtask performance will degrade as a function of conditions which fail to support communication.

The MAPPS model provides information at varying degrees of granularity. The user may request information in one, several or all of the following categories: subtask results (information about each subtask, each time it is simulated during the first iteration), shift results (for each shift during the first iteration, summary information is provided), iteration results (summarized information for the first 5 iterations of a run), and run/task results (summarized information over all iterations of a task). Table 1 presents the detail of the content of each output type. The broad selection of output data provides the MAPPS user/analyst with an efficient and effective tool for gaining greater insight into the performance of selected maintenance activities.

Sensitivity Testing of MAPPS

Upon the completion of major model development efforts and prior to the formal evaluation of the model, MAPPS was subjected to sensitivity testing. The purpose of these tests were: (1) to determine the reasonableness of the model and to ensure proper directionality, and (2) to identify portions of the model which might require calibration prior to formal model evaluation efforts.

The task simulated in the sensitivity tests was "test and repair of a control rod drive motor." This task is performed by two maintainers and is composed of 32 subtasks. In total, 37 sensitivity test runs were completed. Thirty of these runs involved unitary variation of parameters while the remaining 7 addressed simultaneous multiple variation of parameters that allowed the joint effects on model output to be examined with respect to "favorable" and "unfavorable" conditions. Although it is not possible in this paper to discuss in detail all results of the sensitivity testing, an example of the types of results obtained is presented in Table 2. This table outlines the variations in certain output data as a result of varying supervisors acceptance level from .70 (moderately critical supervisor) to .98 (highly critical supervisor). These results, like most of the others obtained during the sensitivity testing indicate proper directionality and plausible changes in magnitude. Results of the sensitivity testing indicated some need for minor calibration within the model which was subsequently carried out. Detailed results of the sensitivity testing is presented in Ref. 8.

MAPPS Model Evaluation

The implementation, ultimate widespread use and overall value of a model such as MAPPS is dependent upon a number of critical issues. In order to ensure that MAPPS meets the goals for which it was designed and in order to gain a broader perspective of its potential applications, a comprehensive effort focused upon model evaluation has been undertaken. These currently on-going efforts specifically address critical issues within four general areas: practicality, acceptability, usefulness and validity. Practicality refers to such issues as the feasibility of implementing MAPPS on a computer installation other than that on which it was developed (i.e., transportability), the cost of implementing the model, ease of developing input data and training requirements for potential users. Acceptability is reflected in the users attitudes toward MAPPS and the information it produces, i.e., the sum of positive and negative responses to MAPPS and its characteristics. Usefulness refers to the compatability of the model's diverse outputs and capabilities with the needs of its various potential user categories (i.e., NRC personnel, NPP maintenance management, architect/engineers), along with its convenience and economy as an analytic tool. Validity refers to the extent to which: (1) the simulation is internally consistent (internal validity), and (2) predicts NPP maintainer performance (empirical validity).

The assessment of internal validity refers to determining the consistency and coherence of the MAPPS logic design; i.e., it is an examination of the relations within the set of MAPPS variables and whether these variables behave logically and consistently.

The assessment of empirical validity within the current study is subdivided into an indirect and direct approach. Within the indirect approach, the issue addressed is the correspondence of a consensus of expert opinion regarding likely maintenance performance under specified conditions with the projections of MAPPS. Within the direct approach, measures of actual NPP maintenance performance will be obtained and the conditions surrounding that performance will be measured. Results of the measures obtained in the field will be correlated to similar measures generated by MAPPS. Table 3 lists the various evaluation issues that are being addressed.

Practicality issues focus primarily on characteristics that are obtainable directly from the model developers (Applied Psychological Services, Inc.). Their expert judgement, prior experience, and close association with the MAPPS model will provide qualitative and quantitative elaborations of the identified issues of interest. Two other issues (expandability and generality) will be addressed in a similar manner.

Issues pertaining to the reaction of the various user groups to the model are being assessed using a case approach method. This method, as applied in the MAPPS evaluation efforts consists of presenting to each user group a set of three problem scenarios developed around typical decision-making situations that may be encountered by each user group. Each user group is then supplied with output from MAPPS, applicable to the current situation and is asked to respond to a number of questions related to the usefulness of the MAPPS data in reaching a decision. A typical problem situation would include information pertaining to personnel, the environment, procedures, equipment accessibility, protective clothing, etc., and would also pose a particular problem. For example, an I&C supervisor wants to decrease performance time without increasing errors. He suspects that heat, noise level and procedures quality may affect performance time and errors. In addition he suspects that the infrequent performance of this task and possible lack of training may also affect performance. His concern is to what degree do each of these factors contribute to decreased performance time and errors and which problem areas to address initially to obtain maximum improvement.

After being presented with this problem situation, each user group will be presented with output from MAPPS and asked to respond to questions such as:

- For the decision at hand, how useful is the model in supplying needed information?

- For the decision at hand, how would you use the information supplied by the model?

- For the decision at hand, is the information provided by MAPPS at a sufficient level of detail?

Efforts focused on the internal consistency (internal validity) issue will attempt to verify the coherence and consistency of the MAPPS logic. These efforts will include an examination of the defined relationships within the set of variables included in the MAPPS model and whether the model behaves, in fact, in consonance with them. A correlational approach will be used to examine both the input to module relationships and the module to output relationships. The goal of these efforts are to verify that the behavior of MAPPS is internally consistent with the design intent.

As mentioned earlier, the external validity issue is being addressed by both an expert consensus and empirical approach. For the expert consensus approach, an assessment of the degree of agreement of MAPPS predictions with the predictions of a panel of experts will be made. Three tasks will be developed for presentation to panels of experts from the various user groups and for MAPPS runs.

Each of the three consensus tasks will be described in detail to the panels. The conditions surrounding the performance of each task will be clearly specified and the sequence of subtasks will be presented. Participants will be asked to provide estimates of a number of performance measures such as average task duration time and probability of success. These estimates will be made by each member of the panel on an individual basis. After individual estimates or projections have been completed, the results will be reviewed with the panel collectively. The MAPPS output for the tasks addressed will be presented to the panel, and after a group discussion of results, panel members will be asked individually to reestimate or to revise their original estimates, if they wish. After this second round of estimates has been completed, individual estimates will again be reviewed by the panel. It is expected that after two or three rounds of this procedure, the individual expert estimates will converge and a best estimate of the group opinion will have been obtained.

For each variable judged by the experts, a point estimate will exist. For each model run and variable of interest, a mean and standard deviation will have been generated by MAPPS. This allows direct comparison of the MAPPS results with the opinions of the experts. Agreement between the MAPPS output and the consensus of the experts will be assumed to have been demonstrated for a variable if the expert consensus falls within plus or minus one standard deviation of the MAPPS mean.

For the empirical approach to the external validity issue, the performance of a set of actual maintenance tasks in NPP settings are being observed and performance measures are being obtained. Analogous MAPPS simulations will be completed, and correlational comparisons will be made across the observed and the MAPPS output data. The collection of field data at one NPP has been completed and collection of data at a second plant is scheduled for late FY-1984.

The tasks addressed at the first NPP involved two-man teams of either instrument and control technicians or electricians, depending on the task to be observed. Ten teams of maintainers were observed for each task and data were collected on a number of performance measures. The tasks addressed were: (1) corrective maintenance of Limitorque valve actuators, (2) source range channel calibration, and (3) reactor pressurizer channel wide range level calibration. The performance measures observed included task completion time, number of subtask successes/failures, the number of undetected errors, the number of detected errors, time spent in subtask repetition and waiting time. A maintenance supervisor assisted the data collection team in collecting data and in identifying errors committed by the observed teams.

Following the observation of the performance by each team, interviews with the team were conducted and paper and pencil questionnaires administered to determine various maintainer characteristics that would be necessary for completing the corresponding MAPPS run. Characteristics determined in this manner included psycho-motor ability, intellective ability, aspiration level, and stress threshold.

Preliminary results from the comparison of observed task times to task times predicted by MAPPS shows very good agreement. For task 2, the observed mean time and standard deviation were respectively 73.6 and 14.2 minutes. The MAPPS model predicted 78.0 and 13.4 minutes, respectively. For task 3, the observed values were 119.8 and 17.6, respectively, while MAPPS estimated values of 117.0 and 9.6, respectively. Although these results should be regarded as preliminary in nature, they indicate the potential for demonstrating a high degree of external validity for MAPPS.

The collection of all necessary MAPPS model evaluation data is scheduled to be completed by the end of FY-1984. Analysis of all data and resolution of the evaluation issues identified in Table 3 is scheduled to be completed by the end of FY-1984. A NUREG/CR report addressing the model evaluation phase of the work in this program will be published by mid FY-1985.

Potential Applications of MAPPS

The primary purpose of the MAPPS model is to provide input (in the form of human reliability data) to the human reliability analysis portion of PRA. Chapter 4 of the PRA Procedures Guide[9] clearly recognizes the potential of human errors in contributing to the overall risk associated with the operation of a NPP. Effective application of a human reliability analysis in support of PRA studies requires a comprehensive source of human reliability data. One of the primary sources for such data is the Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications.[10] The human errors dealt with in this "handbook," however, are primarily NPP operator-oriented and generally cannot be applied to NPP maintenance tasks. The MAPPS computer simulation model is capable of supplying to the human reliability analyst a source of data specifically geared toward the NPP maintenance context. Although the output data from MAPPS is not in itself empirical, the model is firmly based on NPP maintainer job analytic data, was sensitivity tested against maintenance task analytic data, and is being empirically validated against observed (not historic) data from the field. Thus, MAPPS should represent a rich and reliable source of data for the PRA analyst.

Data from MAPPS with respect to maintainer tasks can be used within the human reliability analysis portion of PRA in the same way that the "handbook" data pertaining to operator tasks is currently used. One distinct advantage of the simulation technique over the THERP (Technique for Human Error Rate Prediction) technique as employed in current human reliability analysis methods is that many of the subjective aspects of the quantitative assessment phase of these methods are addressed in the simulation itself. Manual analyst tasks such as assigning human-error probabilities, estimating the relative effects of performance shaping factors, assessing dependence, determining success and failure probabilities and determining the effects of recovery factors are addressed automatically by the simulation. Although it is cautioned that no methodology or technique should be indiscriminately applied, proper application of the MAPPS model should lead to a source of maintainer data for the analyst that is comprehensive, reliable, and relatively free from potential subjective bias by the analyst.

As illustrated in Table 1, MAPPS has the capability of generating a large number of output variables at four hierarchial levels. These data provide the analyst not only with an overall measure of task performance such as probability of task success, but also with a host of other measures which provide significant insight into the task being addressed. Because of this, the application of the MAPPS model extends beyond its primary purpose for PRA. Some simple but relevant applications of MAPPS are listed:

- Aid in determining optimal manning requirements for a series of critical maintenance tasks.

- Aid in determining optimal scheduling of maintenance tasks given a fixed work team.

- For a given set of maintenance tasks, determination of abilities required and degree of familiarity required for successful performance.

- Determination of the degree of performance improvement of newly written procedures compared to old procedures.

- Given a specific budget, determination of whether increased training, enhanced procedures or better environmental control would be most cost effective in increasing performance.

- Given two system designs of equipment, determination of which is easier/most cost effective to maintain?

- Given two different maintenance structures at two similar NPPs, determination of which structure is best, and why?

- Given a new system design, determination of how much downtime should be expected due to scheduled maintenance?

- Given a new maintenance procedure, determination of what part is most difficult/most stressful/most error prone for maintainers?

Other specific uses of the information provided by MAPPS include but are not limited to: (1) maintenance system design evaluation (e.g., estimating time to repair existing systems, identifying maintainability problems in existing systems, evaluating maintenance procedures), (2) maintenance operations analysis (e.g., comparison and optimization of maintenance strategies, maintenance planning/scheduling), and (3) contributing data for a human factors data store.

Summary and Conclusions

The development of the MAPPS model has been completed and the model is currently undergoing evaluation. These efforts are addressing a number of identified issues concerning practicality, acceptability, usefulness, and validity. Preliminary analysis of the evaluation data that has been collected indicates that MAPPS will provide comprehensive and reliable data for PRA purposes and for a number of other applications.

The MAPPS computer simulation model provides the user with a sophisticated tool for gaining insights into tasks performed by NPP maintenance personnel. Its wide variety of input parameters and output data makes it extremely flexible for application to a number of diverse applications. With the demonstration of favorable model evaluation results, the MAPPS model will represent a valuable source of NPP maintainer reliability data and provide PRA studies with a source of data on maintainers that has previously not existed.

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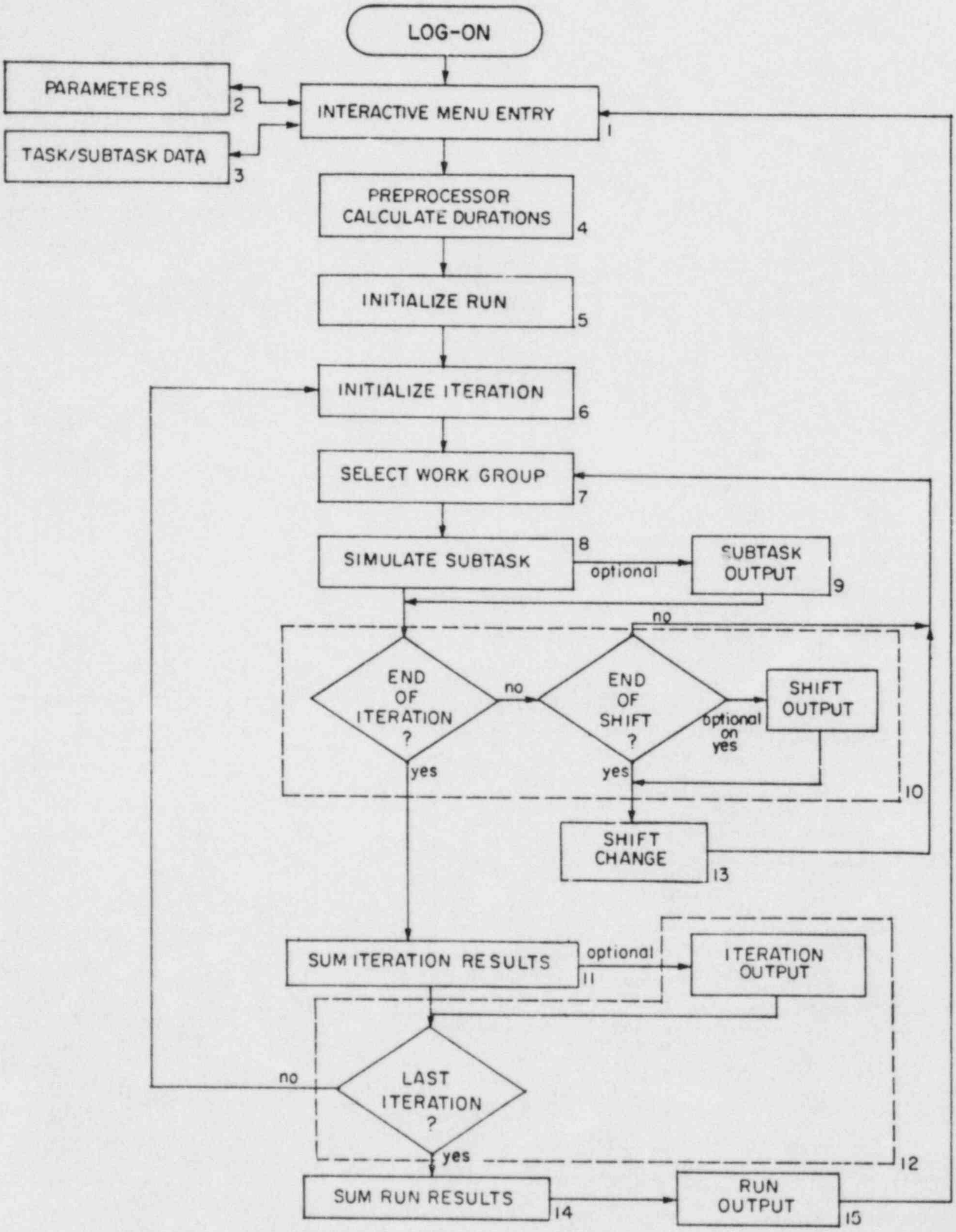

Figure 1. MAPPS functional processing.

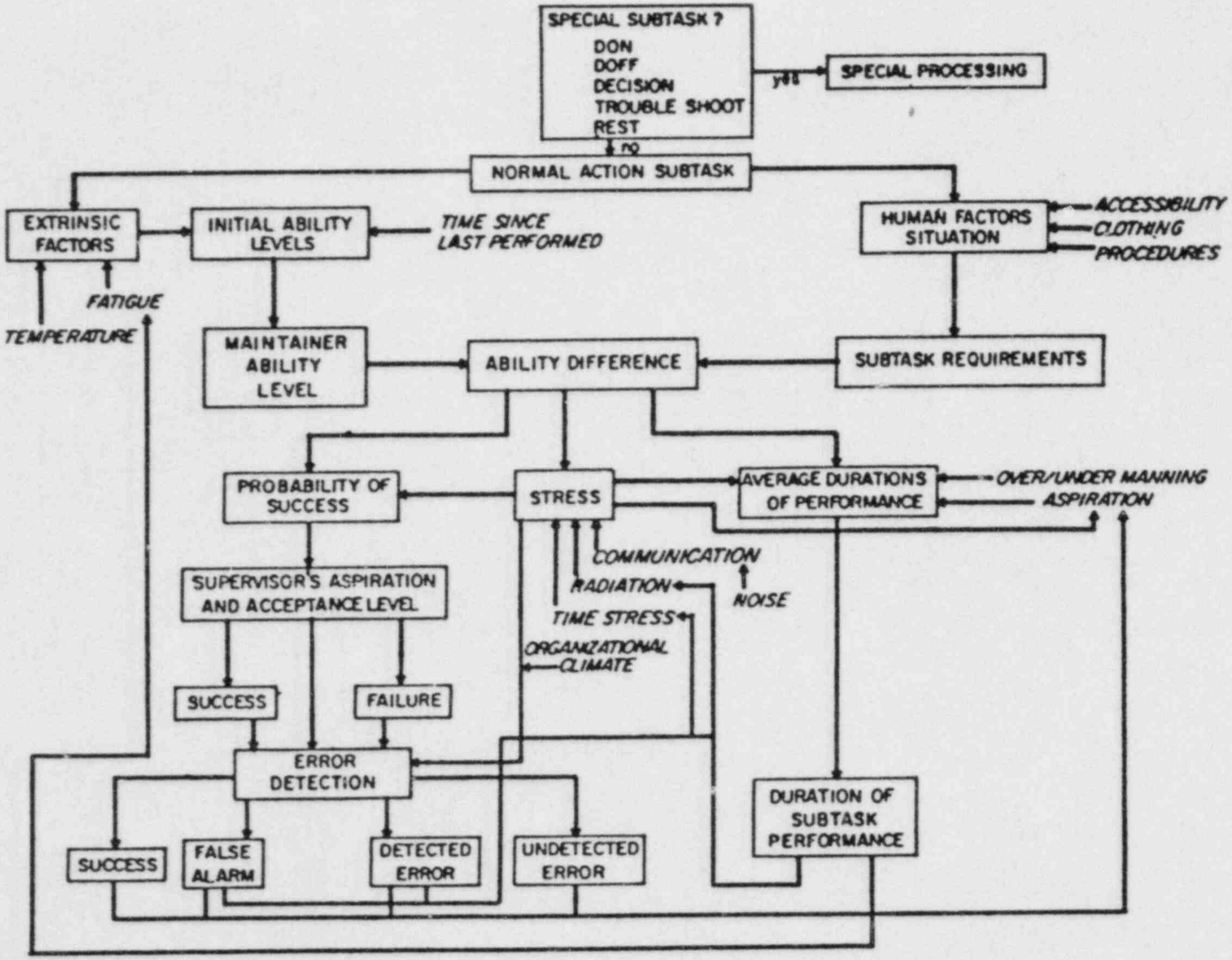

Figure 2. Overview of logic flow within MAPPS model.

Table 1. Detail of Content of Each Output Type

Output	Output Type: Subtask	Shift	Iteration	Run (Task) Summary
General Information				
Subtask Number	X		X	X
Subtask Type Number	X			
Subtask Description	X		X	X
Task Number		X	X	X
Task Description		X	X	X
Iteration Number			X	X
Number of Iterations				X
Shift Number		X		
Reason for Shift Change		X		
Run Identifier			X	X
Run Date			X	X
Source of Task Analysis				X
Subtask Performance				
Attempts	X		X	X
Outcome-Success	X		X	X
Outcome-Detected Error	X		X	X
Outcome-Undetected Error	X		X	X
Outcome-False Alarm	X		X	X
Outcome-Ignore	X		X	X
Probability of Success	X		X	X
Start Time	X			
End Time	X	X	X	X
Work Duration	X		X	X
Wait Duration	X			
Accessibility Effect	X			
Procedures Effect	X			
Last Subtask Performed		X	X	X
Task Performance				
Outcome			X	X
Performance			X	X
Effectiveness			X	X
Error Detection Ratio			X	X
Duration			X	X
Productivity			X	X
Error Consequence Index			X	X
Duration			X	X
Time Overrun/Underrun			X	X
Time Spent in Repeats			X	X
Emergency Duration			X	X
Subtask Preceding Emergency			X	X

Output	Output Type: Subtask	Shift	Iteration	Run (Task) Summary
Characteristics by Maintainer Type				
Type			X	X
Number			X	X
Work Time			X	X
Wait Time			X	X
Rest Time			X	X
Outcome-Successes			X	X
Outcome-Detected Errors			X	X
Outcome-Undetected Errors			X	X
Outcome-False Alarms			X	X
Outcome-Ignores			X	X
Personnel Shift Change Information				
Maintainer Type		X		
Personnel Replaced		X		
End Ability Level-Intellective		X		
End Ability Level-Perceptual-motor		X		
Radiation Absorption		X		
Time on Task		X		
Personnel Characteristics				
Ability Level-Intellective	X		X	X
Ability Level-Perceptual-motor	X		X	X
Ability Difference-Intellective	X			
Ability Difference-Perceptual-motor	X			
Ability Difference Effect	X			
Fatigue Effect-Intellective	X			
Fatigue Effect-Perceptual-motor	X			
Heat Effect-Intellective	X			
Heat Effect-Perceptual-motor	X			
Pace Adjustment Factor	X			
Time Stress	X		X	X
Communication Stress	X			
Total Stress	X		X	X
Maximum Total Stress			X	X
Subtask with Maximum Stress			X	X
End Total Stress		X	X	X
Number of Maintainers	X			
Personnel Ratio	X			

TABLE 2

Results of the Unitary Variation of Supervisors Acceptance Level from Moderately Critical (.70) to Highly Critical (.98)

A. Duration time increased by 37 minutes for this 6-7 hour task.

B. Success proportion decreased by 24%.

C. Failures due to time overrun increased by 12%.

D. Failures due to excessive repetition of failed subtasks increased by 12%.

E. Maximum stress of the work crew increased by 18%.

F. The performance index (ratio of the number of successfully completed subtasks across all iterations to the total number of subtasks attempted) decreased by 11%.

TABLE 3

Model Evaluation Issues

Practicality Issues

Cost of Ownership
Personnel Requirements to Run MAPPS
Training Required to Run MAPPS
Maintenance Requirements of the Code
Hardware Requirements of MAPPS
Portability and Satellite Requirements
Compatability with Other Computer Systems
MAPPS Run Requirements (i.e., data required)

Acceptability and Perceived Usefulness Issues

Reaction of NRC
Reaction of Utilities
Reaction of Architect/Engineer Firms
Expandability
Generality
Completeness

Validity Issues

Internal Consistency
External Validity

ESTIMATION OF HUMAN ERROR PROBABILITIES USING LICENSEE EVENT REPORTS*

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The purpose of this paper is to describe a method developed at Brookhaven National Laboratory for estimating Human Error Probabilities (HEPs) using field data from operational experience at nuclear power plants. This is one of four general areas of NRC-sponsored research aimed at developing HEPs for use in Probabilistic Risk Assessment (PRA): (1) structured expert judgment, (2) analysis of training data, (3) performance modeling, and (4) analysis of operational field data. Field data are any data obtained from actual experience in operating plants. An example of a good source of field data is the Licensee Event Report (LER) file which contains standardized reports of events from licensees operating nuclear power plants. They must be filed for a set of defined occurrences as a condition of maintaining an Operating License for a NRC-licensed reactor facility. Because many contain information on human errors, these reports can be used as a source of data in the process of estimating the likelihood of similar human errors occurring. The method presented in this paper was developed to use LER data, but with minor adjustments can be used to accommodate any good field data.

To estimate HEPs using any field data a "rate" for specific errors can be first estimated by counting the number of specific types of errors that occurred and then estimating the number of opportunities for each type of error counted. The number of errors can be divided by the number of opportunities for those errors to derive a rate:

$$\text{Error Rate} = \frac{\text{Number of human errors of a specific type}}{\text{Number of opportunities for that type of error}}$$

---

*This work was performed under the auspices of the U.S. Nuclear Regulatory Commission and is documented in NUREG/CR-3519 by K. J. Voska and J. N. O'Brien.

Rates developed in this manner are a measure of how often a specific human error occurred during a certain number of opportunities. That rate can be reduced to estimate the likelihood that a similar error will occur given a single opportunity which is the form useful in PRA. That is:

$$\frac{\text{Number of errors of a specific type}}{\text{Number of opportunities for that type of error}} = \frac{\text{Likelihood of an error}}{\text{A single opportunity for that type of error}}$$

In order to determine the rate of any human error's occurrence, first the total number of human errors of a particular type must be counted. PRA practitioners need HEPs for particular errors on particular systems or components and often there is only one error of a given type due to the specificity of HEPs needed for PRA. When possible, errors of a specific type should be aggregated to minimize uncertainties. If only one error is used, the existence of a second, uncounted error would double the rate if counted. However, if several errors are counted, the rate which would result from including an additional uncounted error would not be substantially different. As a result, more confidence can be placed in HEPs derived from field data for particular errors that have been reported several or more times.

In attempting to analyze the complete population of human errors described in the LER file, a problem arises because many events reported in LERs are not explicitly identified as having been caused by human errors. This may be a problem with other types of field data as well. For example, a pump failure may be attributable to overtorquing of a critical bolt during maintenance, but be explicitly identified in the LER as an equipment failure rather than a human error. As a result, LERs must be reviewed individually to identify implicit human errors which may be associated with reported equipment failures. Two reports,[1] detailing such an effort, reviewed almost 12,000 LER abstracts to identify implicit human errors. This analysis expanded the percentage of human error-related LERs from approximately 2% of all LERs explicitly attributed to human error, to 9% when implicit human error causes were also identified and counted. These reports presented a system level method of HEP estimation using LERs, but PRAs also require subsystem, element, and component level HEPs. The method described in this paper is meant to accommodate these needs.

The LER-HEP method described in this paper is meant to be objective, structured, and stand-alone. It was developed in such a way as to make it useful to PRA practitioners, as well as being a general purpose method, without requiring extensive experience or familiarity on the part of the analyst in analyzing LERs. Other types of field data can be used if available.

Part I of the LER-HEP method is aimed at identifying the specific type of human error reported in an LER containing such information. Figure 1 presents the worksheet for Part I. In order to keep this method consistent with NRC's general research on human reliability, error descriptors are tied to those used to classify HEPs in the NRC Human Reliability Data Bank which is currently under development and is meant to support PRA needs generally.[2] Part I also contains a description of LER formats and provides instructions for reviewing and recording all relevant information from them. Each entry on each worksheet is fully explained in NUREG/CR-3519.

Once the relevant data have been recorded in Part I, Part II of the method provides a technique for estimating the number of opportunities which occurred for the specific type of error identified. When the particular human activity associated with the error identified in Part I is routine and periodically performed (e.g., monthly testing, pre-startup routines), task analyses and plant procedures can be used to systematically determine opportunities for that error. If the error reported is not related to routine and periodic activities (e.g., inadvertent activation of a safety-related system), then opportunities can be estimated using expert judgment techniques.[3] By surveying a sample of LERs, it was determined that approximately 20% of all human errors implicitly and explicitly reported in LERs can be subject to systematic determination of opportunities for error because of the routine and period nature of associated activities. The rest can be subject to expert judgment techniques. Figure 2 is the worksheet for Part II of the LER-HEP method.

Part III is used to combine counts of specific human errors with opportunities for those specific errors in order to estimate an error rate. Figure 3 is the worksheet for Part III. A sufficient period of time must be surveyed so that a number of human errors of a specific type can be counted. That number is then divided by the total number of opportunities for that error occurring during that same period of time. The resulting rate is then estimated to be equal to the likelihood of the error given one opportunity (i.e., an HEP for that specific human error).

The report goes on to assess the practicality, acceptability, and usefulness of the method. As matters of practicality, the availability of LERs and the required logistics and support necessary to administer a centralized effort to analyze past and current LERs were examined. Acceptability of the LER-HEP method was assessed using a survey of PRA practitioners, the results of which indicated a willingness to employ such a method in performing PRAs. Usefulness was assessed by examining the compatability of HEPs derived from this method with PRA data banks and methods. This assessment concludes that the LER-HEP Method can be used to generate HEPs from field data to support PRA activities in a cost effective and objective manner.

Recommendations are made to further improve the effectiveness of the method and develop its application to other field data. It is recommended that:

1. NRC consider the feasibility of integrating aspects of the LER-HEP method into the Sequence Coding and Search System being developed by NRC to analyze the LER file.

2. The LER reporting form, used by licensees under NRC requirements, be amended to include an entry for opportunities when human error is involved.

3. NRC consider the feasibility of using an anonymous reporting system so that human errors do not go unreported or reported as equipment failures in order to avoid punitive actions.

4. NRC con:ider direct observation of human activities in plants to validate important opportunity estimates.

5. NRC establish a communication link between NRC R analysts and those individuals filing LERs in order to better document necessary information.

6. PRA-HEP needs to be clearly explicated in terms of what and specific types of errors are most important to generate.

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Figure 1

HEP WORK SHEET PART I

Basic Information	Docket:		1.
	LER Number: 2.	Control Number:	3.
	Plant Name: 4.	Sister Dockets:	5.
	Plant Type: 6.	Reactor Vendor:	7.
	Event Date: 8.	Report Date:	9.
Error Information	Error Date:	Source:	10.
	Number of Identical Errors:		11.
	Number of Dissimilar Errors: 12.	Page Number:	13.
	Periodic:	Source:	14.
	Error Type:	Source:	15.
	Error Description:	Source:	16.

		Descriptor	Source
Human Actions	Position:		17.
	Duty Area:		18.
	Task:		19.
	Task Element:		20.

		Descriptor	Source
Equipment Characteristics	System:		21.
	Subsystem:		22.
	Component:		23.
	Element:		24.

Interface	Interface Level:		25.
Performance Shaping Factors		Weight:	26.
		Weight:	27.
		Weight:	28.
Summary			29.

Figure 2

HEP WORK SHEET PART II

Page Number:	Number of Unique Classes:

	Activity	Similar Human Actions	Annual Repetition	Source
Inventory of Human Actions				
	Total Annual Repetition			

	Identical Equipment	Quantity	Source
Equipment Inventory			
	Total Similar Equipment		

Opportunity

Total Annual Repetition [ ] × Total Similar Equipment [ ] = Total Opportunity (this Page) [ ]

Total Opportunity (other Pages) → + [ ]

Total Annual Opportunity [ ]

Figure 3

HEP WORK SHEET PART III				
Error Classification	Position: 1.		System: 5.	
	Duty Area: 2.		Subsystem: 6.	
	Task: 3.		Component: 7.	
	Task Element: 4.		Element: 8.	
	Interface Level: 9.			
Survey Period	Survey Begin: 10.		Survey End: 11.	
	Survey Length, Years: Months:		Decimal: 12	
Plant Information	Plant	Factor	Plant	Factor
	13.			
Total Opportunity	Total No. of Plants: 14.		Average Factor: 15.	
	Total Plant Years: 16.		Total Operation Time: 17.	
	Total Annual Opportunity: 18.		Total Opportunity: 19.	
Human Error List	LER Number	Errors	LER Number	Errors
	20.			
	Total Number of Errors: 21.			
Human Error Probability	$\frac{\text{Total Number of Errors (Line 21)}}{\text{Total Opportunity (Line 19)}} = \frac{22.}{23.}$		24.	

NUCLEAR POWER SAFETY REPORTING SYSTEM
IMPLEMENTATION PLAN, CONCEPT EVALUATION & OPERABILITY TEST

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1. INTRODUCTION

During the 1984 fiscal year, the Aerospace Corporation continued to assist the U.S. Nuclear Regulatory Commission (NRC) in evaluating the concept of a Nuclear Power Safety Reporting System (NPSRS). The NPSRS concept embodies a voluntary, nonpunitive, third party managed human factors data gathering system that (with the NRC as its parent agency) could be used for identifying and quantifying factors that contribute to the occurrence of safety problems involving personnel in nuclear power plants. NPSRS data could be used to: (1) support efforts to quantify the human reliability elements of probabilistic risk assessments (PRA's); (2) to evaluate the influence of various nuclear power plant systems on human error-proneness within the system; and (3) to aid in the development of design criteria for human-machine safety systems.

In 1983 the Aerospace Corporation and NRC examined the Federal Aviation Administration (FAA) supported, and National Aeronautics and Space Administration (NASA) administered, Aviation Safety Reporting System (ASRS) to determine its applicability as a model for collecting data on human factors related incidents in the nuclear industry. It was determined that the ASRS presented a viable model for many of the fundamental attributes of an NPSRS (Ref. 1). The basic elements and requirements of an NPSRS were then identified and defined (Ref. 2). The objectives of this year's studies have been to prepare a draft of a comprehensive package of NPSRS concept implementation plans and procedures (Ref. 3), to develop a plan for evaluating the concept and performing an operability test to assess the practicality, acceptability, and utility of the NPSRS concept (Ref. 4), and to initiate the work needed to support the plans for system evaluation.

2. IMPLEMENTATION PLAN

Pursuant to assessing the practicality, usefulness, and acceptability of the proposed NPSRS, a draft implementation plan was developed (Ref. 3). An objective of this effort was to provide a vehicle for more detailed examination of the feasibility of such a system than was possible in the earlier portions of the study. The elements of the system, its requirements, specifications, detailed procedures, process flow, manpower, and other resource requirements were presented in the implementation plan, along with a taxonomy for implementing the data collection and analyses portion of the system. In order to provide a first order example of how the system implementation might proceed through utilization of its basic forms and procedures, a simulated report was presented together with a description of how such a report would be processed. The creation of this functional and operational description of the NPSRS made possible a more definitive evaluation of the basic NPSRS concept, provided more insight into the remaining unanswered issues associated with the system, and also laid the foundation for an operability demonstration test and other procedures for addressing the remaining system issues.

3. OPERABILITY DEMONSTRATION TEST

A prototypical test of the implementation of the NPSRS, of limited duration, involving mock analysts, mock reporters, and quasi-ficticious reports is scheduled for performance this year. Active and retired nuclear power plant operators are being recruited to function as mock reporters. Experienced nuclear power plant personnel are also being recruited to serve as mock analysts for demonstrating the effectiveness of reporting and analysis procedures.

It is anticipated that a number of critical questions will be resolved when the operability demonstration test is completed. Among these are questions regarding the adequacy of the NPSRS forms and procedures as a medium for information flow through the system, questions regarding the usefulness of the output data for its intended purposes, and questions pertaining to the acceptability of the proposed system to persons within the nuclear community.

The adequacy of the forms and procedures will be determined by having the participants of the demonstration test use them to implement the test program. Observations of the tests, interviews with the test participants and examinations of their output products will be used to establish the sufficiency of the implementation plan in this regard.

The usefulness of the data resulting from the operability demonstration test will also be assessed. The data will be examined to determine its suitability for input into other human factors data bases and to determine if it satisfies its other intended purposes. Test participants will also be interviewed to determine how useful they believe the data to be and to determine the basis for their opinions.

In addition to questions regarding the practicality and usefulness of the NPSRS, questions pertaining to its acceptability to members of the nuclear community must also be addressed. A series of structured interviews regarding the NPSRS are an integral part of the Concept Assessment portion of the plan. Management and operational personnel from utilities operating nuclear power plants, staff members from the NRC, personnel from other federal safety reporting systems, and the test participants will all be requested to participate in a series of structured interviews regarding the NPSRS.

4. 1984 FISCAL YEAR RESULTS

In addition to having published both the draft Implementation Plan and a description of the Operability Demonstration Test, considerable progress has been made toward initiating the actual test itself. A number of prospective sources for demonstration test participants have been contacted and several viable candidate organizations have been identified. In connection with the Concept Assessment portion of the plan, the process of setting up interviews with interested parties has also been initiated and in some instances completed. A number of utilities have agreed to support the program as participants in a survey concerning the issues associated with the NPSRS. Management personnel with nuclear power plant responsibilities, and at some locations nuclear power plant operators, have been designated to respond to the structured interviews used in the survey. Staff members at the NRC that are to be interviewed have been identified, as have personnel from other federal safety reporting systems.

A protocol for the structured interview has been developed and interviews with officials from the Aviation Safety Reporting System (ASRS) have already been completed. Ten general areas are covered in the interview, which are as follows:

1. Procedures for Initial Implementation of the System
2. The Identification of Acceptable Reporting Groups
3. Conditions of Immunity
4. Criteria for Minimum Content for Report Acceptability
5. The Makeup and Function of an NPSRS Advisory Committee
6. Third Party Manager Related Issues
7. Anonymity Protection Methods
8. Data Utility for Users
9. Concept Acceptability for Users
10. Concept Acceptability for Reporters.

The issues associated with Procedures for Initial Implementation of the System include such things as defining how responsibility for the system is to be transferred from the NRC to the third party agency that will administer the program, and related questions regarding how to provide initial publicity for the system, how to provide initial services, how to motivate people to use the system, how to insure the continued viability of the system, how to establish feedback mechanisms, and how to define the NRC's role as the parent agency of the system.

In the case of the ASRS, the program is supported financially by the FAA and administered by NASA. NASA in turn contracts out the day-to-day operation of the program to the Battelle Corporation. The transfer of authority from the FAA to NASA was easily executed. The terms of the arrangement were stipulated in a memorandum of agreement between the FAA and NASA and the system was defined for users in the FAA Advisory Circular (AC 00-46, March 31, 1976). Immediately thereafter, the FAA ceased its in-house efforts to implement an Air Safety Reporting System and NASA initiated the present ASRS system. The system was publicized through mass mailings to pilots with active medical certificates, and other forms of public announcements aimed at the aviation industry. Initial services consisted of distributing blank report forms and accepting completed reports. People were motivated to use the system primarily because they desired immunity for their actions which might have inadvertently broken FAA regulations. The viability of the program was insured by soliciting and cultivating strong industry based support for the ASRS system. Feedback about the system was solicited during callbacks to reporters and from the advisory committee established by NASA to evaluate the performance of the ASRS. The role of the FAA as the parent agency was strictly defined as that of being a funding source and beneficiary of the data. Other than providing the system's financial support, and having a role as part of the advisory committee, the agency committed itself not to impact the day-to-day operations of the ASRS.

The issue of who should be permitted to submit reports to the NPSRS was brought up with ASRS representatives. Their answer was that anyone in a position to be aware of conditions or situations relevant to nuclear power safety should be permitted to submit reports. Their rationale was that the more obstacles that are placed in the path of potential reporters the less likely they are to use the system. If would be reporters must question whether or not they may be permitted to submit reports, they will be that much less likely to use the system.

Within the aviation industry, individuals are provided one time immunity from regulatory redress for violation of an FAA regulation if they have filed a report with the ASRS describing the incident, with two exceptions. Criminal acts and incidents resulting in accidents are not excused. (The FAA is considering the revision of the once in a lifetime immunity clause to provide immunity from enforcement actions from one violation every five years.) The granting of immunity is viewed as an essential part of the ASRS's success. It is a prime motivator for people to submit reports. There was only one reported abuse of this system, and it took place during the initial period of ASRS operation, when the FAA had extended an unrestricted warranty of immunity to reporters for any (and all) reported incidents. One pilot is reported to have filed five reports in eight months, seeking exemption from redress for bad flying habits. In each case the person received the promised immunity from FAA enforcement procedures, but (as noted above) the rules have been changed to prevent this kind of abuse.

Just as anyone may submit a report to the ASRS, any kind of report is accepted. However, reports reflecting criminal conduct or pertaining to accidents are forwarded to the appropriate authority and no immunity is granted. As might be expected the reports vary considerably in terms of content and quality. Analysts at the ASRS initiate callbacks to many reporters, and the analysts' comments exert a kind of quality control over the data base. If the analyst is dubious about a report or suspects a covert reason for its filing they will so note that in their comments. The ASRS's advice with respect to NPSRS report acceptability was to accept all reports and hire good analysts capable of separating the wheat from the chaff.

The ASRS personnel reported that the NASA advisory committee was an integral and very important and useful part of their system. They advised that for the NPSRS the optimal mix might include industry representatives, manufacturers representatives and representatives from the NRC.

In terms of a third party manager for the NPSRS, the ASRS officials recommended an academic or military type of organization. They felt it essential that the managing organization be neutral, free from economic dependence upon the program, and very knowledgable about the subject.

The proposed report form for the NPSRS was presented to the ASRS respondents, and the suggested methods for protecting reporter identity were described. They were asked to comment upon the sufficiency of the forms and procedures for the purpose of preserving the anonymity of the reporter. The proposed NPSRS methods for achieving this end are almost identical to those employed by the ASRS and were perceived as generally adequate. The ASRS representatives also discussed their experience with methods of maintaining the anonymity of reporters and the integrity of the ASRS data base and meeting the requirements of the freedom of information act.

Not being members of the nuclear community, the ASRS officials were unable to directly assess the likely utility of the system to users of the data, or to evaluate its likely acceptability to users and reporters. In general however, based upon the successes of their own system, they were optimistic that the NPSRS would be useful and acceptable to both its constituencies. The ASRS officials were helpful in commenting upon the technical merit of the NPSRS concept and in suggesting additional uses for the NPSRS data. They pointed out that a substantial fraction of the ASRS data pertains to incidents or near incidents that were averted or made less consequential through the actions of aviation personnel involved. The ASRS data is therefore not merely a catalog of disasters, but is also a repository of information on positive mechanisms for resolving or avoiding unsafe system conditions. In the opinion of FAA and ASRS personnel, the aviation industry has benefited from both kinds of reports, and they expressed a belief that the nuclear community could benefit in a similar fashion from a system like the NPSRS.

In summary, the 1984 fiscal year research is proceeding apace on the NPSRS project. By year's end, it is anticipated that the research completed by the Aerospace Corporation will have provided the NRC with the critical, technical background data needed for making decisions concerning the appropriateness of this type of reporting system for application within the nuclear utility industry.

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PROCESS EVALUATION OF THE HUMAN RELIABILITY DATA BANK

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ABSTRACT

The U.S. Nuclear Regulatory Commission and Sandia National Laboratories have been developing a plan for a human reliability data bank since August 1981. This research is in response to the data needs of the nuclear power industry's probabilistic risk assessment community. The three phases of the program are to: (A) develop the data bank concept, (B) develop an implementation plan and conduct a process evaluation, and (C) assist a sponsor in implementing the data bank. The program is now in Phase B. This paper describes the methods used and the results of the process evaluation. Decisions to be made in the future regarding full-scale implementation will be based, in part, on the outcome of this study.

INTRODUCTION

Since August 1981, the U.S. Nuclear Regulatory Commission (NRC), Sandia National Laboratories (SNL), and General Physics Corporation (GP) have been working on a research project to develop a data bank for human reliability data. The project is in response to the growing needs of the probabilistic risk assessment (PRA) community for human reliability data. The data bank would serve as a centralized coordinating facility that would categorize, aggregate, and store human reliability data and make the data available to the user community.

The data bank development project consists of the following three phases:

- Phase A: Review previous data bank efforts and develop a conceptual framework for a human reliability data bank.

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- Phase B: Develop a set of detailed procedures for implementing the data bank and conduct an evaluation of the data treatment processes.

- Phase C: Assist a sponsor in the implementation of the data bank.

Phase A was completed in December 1982; Phase B was initiated in May 1983. The work performed in Phase A is described in detail in an earlier publication for the Human Factors Society (Comer & Miller, 1983). A brief summary of Phase A is presented here, followed by a description of the work performed in Phase B and a summary of the results found in the evaluation.

SUMMARY OF PHASE A

A review of existing data bases was conducted in the initial stages of the program to (1) ascertain whether human reliability data that could be used in nuclear power PRAs have been collected and stored in existing data bases and (2) determine characteristics of the data bases that might be useful in the design of a new data base. As a result of the review, it was concluded that there were insufficient data in the existing data bases to adequately support nuclear power PRA activities and that a human reliability data bank specifically tailored to nuclear power PRA applications was warranted. This review is described and three of the data bases that were reviewed are reproduced in NUREG/CR-2744, Volume 1 (Topmiller et al., 1982).

With the results of the data base review as a start, a concept for a human reliability data bank was developed. The intent was to provide one central location for human performance data that could be used to perform a human reliability analysis portion of a PRA. Three primary methods of collecting information on what the user community needed in terms of a data bank were employed in Phase A. They were:

- Telephone surveys of potential users working in the PRA area

- In-person interactive session with members of a PRA team who were currently involved in evaluating human reliability

- Two-day peer review session that addressed the concepts developed for the data bank.

Figure 1 is a graphical representation of the system that evolved during the development of the human reliability data bank. The figure shows the system's inputs and outputs and its four major components, which are:

- Administrative Staff, which coordinates the overall operation of the data bank and interacts with the data suppliers

- Human Reliability Analysis Group (HRAG), which processes data and provides technical input to the system

- Review Committee, which provides final approval of products and reviews changes to the data bank operation
- Data Clearinghouse, which interfaces with the data bank users

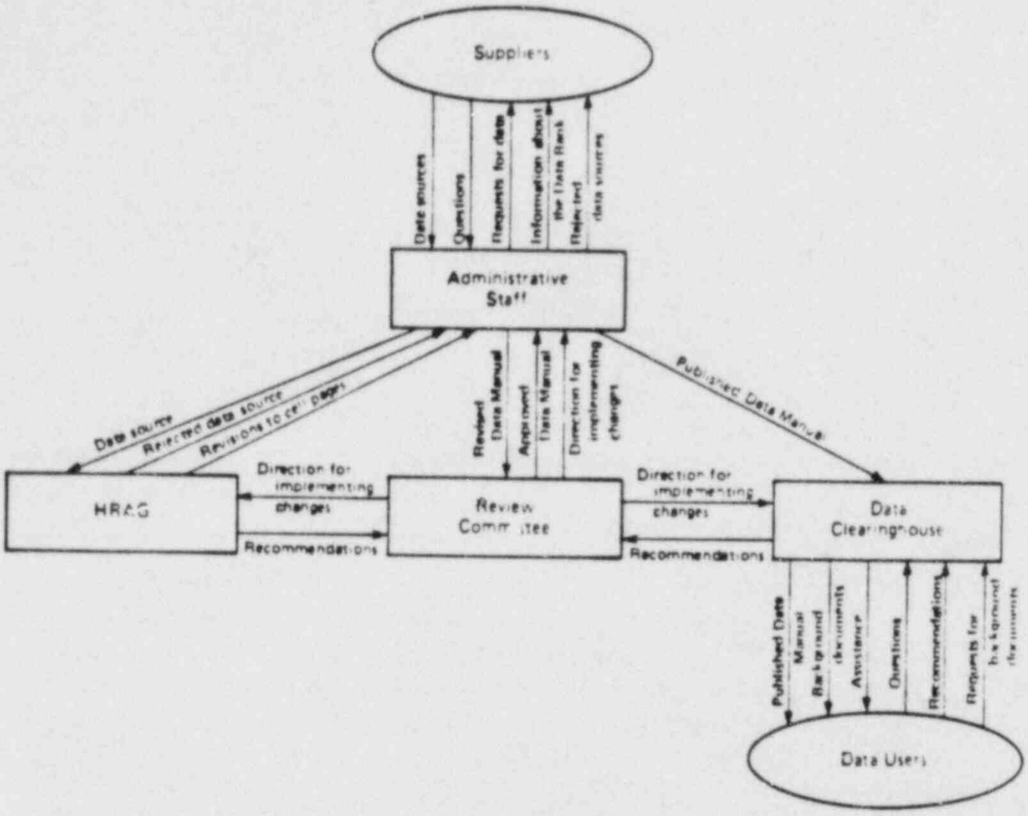

Figure 1. Relationships within the data bank system

The results of the Phase A effort included a concept description for collecting, combining, storing, and retrieving data and a detailed hierarchical data taxonomy. Details on each of these areas are contained in NUREG/CR-2744, Volume 2 (Comer et al., 1983).

PHASE B

Based on the concepts developed in Phase A, Phase B focused on refinement of those concepts through the development of detailed implementation procedures. The work in Phase B also included evaluating the procedures that were developed in terms of practicality, acceptability, and usefulness.

Procedure Development

Early in Phase B, detailed procedures for implementing the data bank according to the system described above were developed. Procedures were developed for processing the data contained in the data bank, reviewing the data manual before it is published, and retrieving data from the data bank. Twelve procedures were developed to address the process of receiving data and entering them into the data bank. A flow chart outlining the sequence of these procedures is shown in Figure 2.

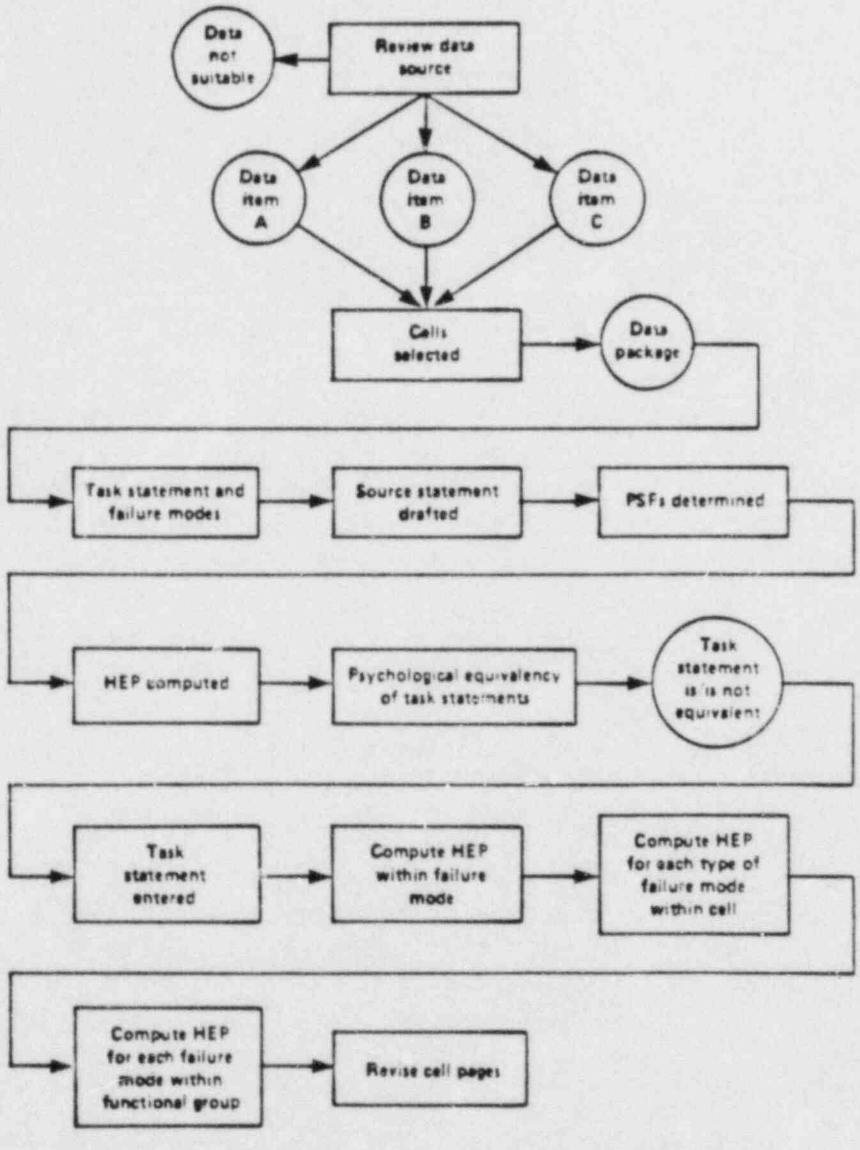

Figure 2. Steps for processing data

Four procedures were developed for revising the data manual, and four additional procedures were developed for retrieving data using the data manual. A "skeleton" data manual (accurate structure with minimal data) was developed to simulate the methods that would be employed by the users to retrieve data. The skeleton data manual contained the following information:

- Introduction (purpose, scope, overview)
- Data treatment (where the data come from and how they are combined)
- Search strategy (how the data bank is organized and how data are located)
- Sample data retrieval problems (examples of how to retrieve data from different levels of the taxonomy and what to do if the desired data are not available)
- Description of the taxonomy (3-level, 16-matrix classification scheme)
- Definitions (every equipment characteristic and human action term used in the taxonomy)
- Data pages (human performance data--this part was fabricated for the purpose of the skeleton data manual)

Evaluation

A second objective of Phase B was to evaluate the data bank in terms of practicality, acceptability, and usefulness. The specific issues that were addressed by the study are:

(1) How much will it cost to implement the data bank?

(2) What personnel resources are required to implement and operate the data bank?

(3) What resources, other than personnel, are required to implement and operate the data bank?

(4) Should the data bank be implemented by the NRC, a national laboratory, or a third party?

(5) Is the data bank structure expandable and adaptable to change?

(6) What qualifications must users have to access the data bank?

(7) What type of organization or individual can provide data to the data bank?

(8) Is the data bank acceptable to the NRC as a means of acquiring human reliability data?

(9) Is the nuclear power industry willing to use data from the data bank?

(10) Are the people working in PRA, HRA, human factors engineering, and general engineering willing to accept the data combination process and to use data from the data bank?

(11) What considerations are involved in the scientific community's acceptance or rejection of the concept?

(12) Are the data contained in the data bank of high quality, i.e., from a valid, reliable source?

(13) Is the data bank compatible with, i.e., responsive to, the requirements of a PRA?

(14) Can the data bank be implemented and operated independently of the developers?

(15) Can data that are retrieved by data bank users be traced back through the data bank process to the original input data?

(16) Can the data bank be used to access data regardless of the analytical technique [e.g., Technique for Human Error Rate Prediction (THERP) or Operator Action Trees (OATS)] used by the HRA analyst?

(17) Will the data clearinghouse be used as a supplement to the data manual or as a replacement for the data manual?

(18) Are the input data requirements feasible; i.e., are they too stringent or too lax given the requirements for PRA data?

(19) What are some additional issues involved in implementing a data bank (e.g., should it be a manual or a computerized system?)

Method

There were four separate methods used to evaluate the data bank. The first two, (1) gathering of data from other data bank systems and (2) conducting an internal analysis of the data bank system, were used to supplement the information obtained from the remainder of the evaluation. The two primary methods used were:

- An in-house evaluation of data processing concepts and procedures
- A mail-out evaluation of the data manual and data retrieval procedures

The three-day, in-house evaluation took place at GP's Columbia, Maryland, headquarters. It involved 15 industry and government participants who represented the disciplines of human factors, PRA/human reliability, and nuclear power plant operations and maintenance. Five teams of these participants simulated the HRAG. The evaluation consisted of group training followed by a series of exercises designed to demonstrate the data processing process, including all 12 procedures. After the group training, the participants divided into teams to perform the exercises under the supervision of a proctor. Each individual worked the exercise alone; then the team worked the exercise together. The membership of each team was changed at the end of each session to lessen the effect of individual influences on team performance. A survey was administered for each exercise. After all 12 procedures had been evaluated, an integrated evaluation was conducted. The participants received the input information for the first procedure and were asked to process it through all 12 procedures. At the conclusion of all the exercises, an interactive discussion was held with all participants to discuss their comments and the issues involved in implementing the data bank system.

The mail-out evaluation of data retrieval involved sending a skeleton data manual to participants along with a series of exercises and a lengthy survey. The exercises contained examples of human reliability problems from previous PRAs with instructions for the participant to work the problems using the skeleton data manual to retrieve hypothetical data. The participants were also asked to use a problem they had worked in the past using the skeleton data manual. The objective was to evaluate the ease of using the data retrieval procedures as well as the responsiveness of the system to PRA needs. Ten government and industry participants who had previous PRA experience were involved in the mail-out evaluation.

Data Analysis

The data analysis involved combining qualitative and some quantitative information from the following sources:

- In-house evaluation--time and accuracy in completion of individual and group exercises, survey comments, interactive discussion notes, and notes kept by proctors regarding questions asked during the exercises.

- Mail-out evaluation--number of correct responses in exercises, survey comments, cover letter comments, and questions phoned in to the data clearinghouse.

Results

Without discussing the quantitative aspects of the results of the evaluation, the following key findings and recommendations resulted from the in-house and mail-out evaluations:

- Teams performed better than individuals on the exercises identified in the implementation procedures as team-oriented.

- Additional resources required do not differ from those previously recommended by GP except for automation of some clerical tasks in the data processing procedures.

- Most participants felt that the data bank implementation should be performed by an organization other than the NRC or national laboratory.

- The data processing tasks should provide better guidelines for chosing appropriate taxonomy levels for input data.

- Most data manual users had little difficulty in locating appropriate data in the taxonomy.

- Most participants felt that the data bank could handle most types of quantitative human error data. A few suggested that this capability be expanded to accept functional and graphical data types.

- A majority of the NRC participants felt that the data bank approach was an acceptable means of acquiring human reliability data.

- Several participants disagreed with the data bank methods for combining data, but none suggested alternatives that would be more satisfactory.

- Most of the participants agreed that the data bank concepts are compatible and responsive to PRA activities.

- Recommendations were made to guard against the acceptance of third-party data in the data screening procedures.

- The data bank was, in general, found to be useful regardless of the reliability analysis technique favored by the user, but some fine tuning may be necessary to improve the technique independence.

- The data clearinghouse was used as a supplement to the skeleton data manual, not as a substitute.

- Most people thought that the input data requirements were reasonable.

SUMMARY AND CONCLUSIONS

The multiyear research program for developing a human reliability data bank has almost concluded. From the beginning, the project has relied heavily on input from government and industry personnel who are involved in PRA and human reliability data. Once all the modifications to the system are made, the NRC will have a specification for a human reliability data bank that is practical, acceptable, and useful for PRA. The NRC will then have the available information to make informed decisions regarding implementation of the human reliability data bank.

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THE INTEGRATION OF HUMAN RELIABILITY ANALYSIS INTO THE PROBABILISTIC RISK ASSESSMENT PROCESS: PHASE 1

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ABSTRACT

The U.S. Nuclear Regulatory Commission and Pacific Northwest Laboratory initiated a research program in 1984 to develop a testable set of analytical procedures for integrating human reliability analysis (HRA) into the probabilistic risk assessment (PRA) process to more adequately assess the overall impact of human performance on risk. In this three phase program, stand-alone HRA/PRA analytic procedures will be developed and field evaluated to provide improved methods, techniques, and models for applying quantitative and qualitative human error data which systematically integrate HRA principles, techniques, and analyses throughout the entire PRA process. Phase 1 of the program

involved analysis of state-of-the-art PRAs to define the structures and processes currently in use in the industry. Phase 2 research will involve developing a new or revised PRA methodology which will enable more efficient regulation of the industry using quantitative or qualitative results of the PRA. Finally, Phase 3 will be to field test those procedures to assure that the results generated by the new methodologies will be usable and acceptable to the NRC. This paper briefly describes the first phase of the program and outlines the second.

SUMMARY AND CONCLUSIONS

Phase 1 of a program to generate a generic PRA process which is capable of regulatory application has been completed. The results of phase 1 of the project include PRA practitioner specific flow charts and specification sheets which summarize state-of-the-art risk methodology as applied to the nuclear power industry. The process has been examined at three levels: a generic level, a practitioner specific level, and the level at which human reliability enters into the process. It was found that the minimum requirements for a PRA includes analysis of seven specific areas, although more extensive assessments have been performed. Also, several different methods have been used to estimate overall plant risk in every aspect of the process.

In order to coordinate the various methods currently in use to provide a data source for NRC use in regulation, phase 2 of the program will involve the creation of a process to resolve issues using a generic PRA process. Using guidance from data collected in phase 1, a testable methodology will be developed which will provide data sufficient to answer questions of regulatory consequence. These methods will be based on current PRA processes, but may be adjusted to meet the needs of the NRC. After this development, the methods will be tested to assure usability and acceptability.

INTRODUCTION

In modeling nuclear power plant (NPP) operations, for probabilistic risk assessment (PRAs), one important consideration has been the inclusion of estimates of human performance and their relationships to nuclear power plant performance. Many techniques have been applied to the problem of plant risk assessment. Some examples of probabilistic risk assessments which were performed previously included the Reactor Safety Study (RSS) and the Interim Reliability Evaluation Program (IREP). However, there has been little, if any attempt to standardize the inputs of human reliability analysis (HRA) into the PRA process, nor has there been a systematic evaluation of the methods currently in use.

As a result of wide variations in PRA techniques, risk assessments in nuclear power plants are often based on different assumptions, and HRA methods, as well as other information about human systems, are treated differently or not at all. Consequently, the use of PRAs for decision making and issue resolution is difficult, at best. Additionally, insights into risk issues may be overlooked because key assumptions and/or data points are difficult to extract or are not present in the current process. The purpose of this three phase research program is to develop and evaluate a set of analytic PRA procedures using a generic set of steps that incorporate specific human reliability analysis techniques and data that specifically address key issues. This is a three phase program delineated as follows:

Phase 1 Analyze published PRAs and guidelines typical of the efforts of major practitioners and describe the various structures currently in use in the industry.

Phase 2 - Develop, stand-alone HRA/PRA analytical procedures using specifications emerging from the Phase 1 research.

Phase 3 - Evaluate the procedures developed during Phase 2.

Phase 1 will be completed in December, 1984. Phase 2 will begin in January, 1985. A detailed description of the work performed in phase 1 and an outline of scheduled Phase 2 work follows.

PHASE 1 - PRA ANALYSIS

In order to define the PRA processes currently in use, a detailed analysis of the best examples of published reports was performed. The reports were those considered the most representative structurally and available to PNL personnel and the practitioners. The PRAs were selected according to two criteria: availability and representativeness of state-of-the-art practice. The initial list of PRAs was selected because they were published, NUREG documents. However, since several PRAs were joint efforts, that is, several practitioners contributed to their completions, analysis of proprietary reports representative of current structure were used by permission of the practitioner. Consequently, names of specific nuclear power generating facilities have been omitted from the following list of major PRA practitioners used in the current study to preserve anonymity. These practitioners were:

1. NUS Corporation
2. Pickard, Lowe, and Garrick
3. Delian
4. Technology for Energy, Corporation
5. Science Applications, Inc.
6. Sandia National Laboratories
7. Idaho National Engineering Laboratories

Of these PRAs, four were Pressurized Water Reactor (PWR) plants and three were Boiling Water Reactor (BWR) plants. Also, six studies were based on reactors operating in the United States while the seventh was a foreign reactor. In the analysis of current PRA practices, the published reports were condensed to reveal underlying process structures at three degrees of detail. These degrees were: a generic degree of detail; a practitioner specific degree of detail; and, the degree of detail where human reliability is included. Each PRA was considered separately, initially, and then was integrated at the generic level. The following discussion describes the results of the PRA analysis.

The Generic Degree of Detail

The purpose of creating a generic PRA structure was to permit the basic comparison of the similarities between and analysis of individual PRAs. Among several candidate taxonomies considered for generic structure, the most useful was one suggested in NUREG-1050 (Draft), "Probabilistic Risk Assessment

(PRA): Status Report and Guidance for Regulatory Application." The flow path suggested in the publication is shown below in Figure 1.

The basic PRA, or level 1 PRA, contains seven generic steps which are completed, in some form by all practitioners. These seven steps, as defined in Figure 1, were as follows:

- initial information collection
- system analysis
  - event-tree development
  - external event analysis
  - system modeling
- analysis of human reliability and procedures
- data-base development
- accident-sequence quantification
- uncertainty analysis
- development and interpretation of results.

IREP studies are level 1 PRAs, as defined by NUREG-1050. A detailed review of this methodology is found in NUREG/CR-2728.

More detailed assessments are defined by level 2 and level 3 PRAs. A level 2 PRA is characterized by the seven previously listed steps, plus containment analysis, which consists of the following parts:

- analysis of physical processes
- analysis of radionuclide release and transport.

The addition of the first part in the containment analysis step allows the analyst to further develop event-trees and create a more complete and realistic

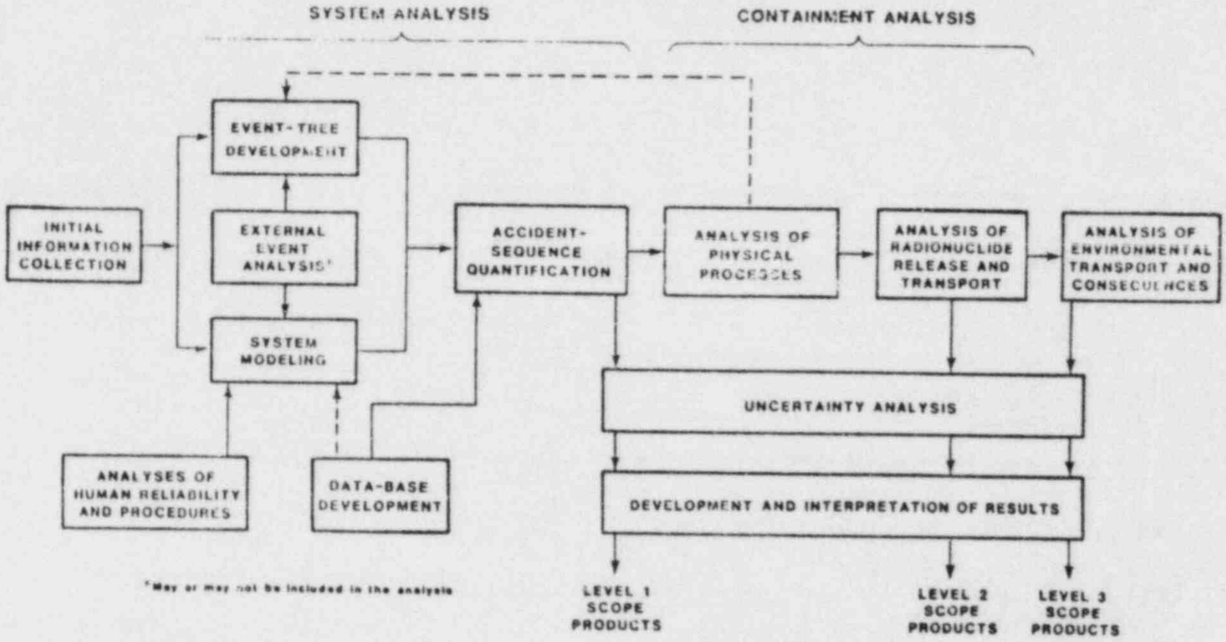

FIGURE 1. Basis for Determining a Taxonomy of Generic Steps

listing of plant evolutions and sequences. The second part in the containment analysis step allows better interpretation of results and analysis of consequences.

A level 3 PRA consists of each of the previous steps plus the following:

- analysis of environmental transport and consequences.

The additional step gives more insight into data interpretation and consequences of potential accident sequences. Quite often, this type of analysis is performed by means of a computer program.

It must be pointed out that the objective for performing the PRA determines its level. A very detailed, level 3, PRA might be performed because the subject plant is near a major population center and the effects of the transport of radionuclides in the environment on public health and economic consequences of the accident are very important. Conversely, a level 1 PRA may be performed on a plant where the objective is merely to identify those aspects of plant design and operation which emphasize sequences leading to core melt. The obvious conclusion is that information gained from a higher level PRA is of a different nature than that of a lower one because of the specific PRA objective and a second aspect, cost.

PRA Practitioner Specifications

After a generic PRA structure was defined, the methods by which practitioners accomplished the individual steps were outlined. This was done by creating a flow chart which represented the RPA process specific to that practitioner. Elements of the flow chart included specific information and knowledge requirements as well as action steps and methods for data manipulation. When available, personnel requirements and other practitioner specific PRA process information was included. The information necessary to complete the flow charts was obtained by reanalyzing the seven PRAs listed earlier. An example of the information flow for one of the PRAs is shown in Figure 2.

Taken as a group, the flow charts represent a collection of the methods used by established PRA practitioners. Specific information regarding various aspects of plant risk is not found within the structure of the flow chart. However, the flow charts indicate where information should be found and a methodology for providing that information. In other words, information which is of a similar type to that obtained by present PRA methodologies may be gained with a slight adjustment to that method. The flow charts also point to information which is unattainable via PRA reports as they are currently structured. It is problematic whether changes would allow the acquisition of this type of data.

HRA Incorporation

Within each step and practitioner specification listed above, information regarding human performance was incorporated in some form in the structure of specification. The form of the input varied depending on the method of modeling human performance. Some HRA techniques use the task analysis based THERP (Technique for Human Error Rate Prediction) approach suggested in NUREG-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications." Another used the OAT method, or Method of Operator-Action Trees, which focuses primarily on errors of response to accident sequences. A third method of human reliability analysis uses a sensitivity analysis task somewhat related to THERP. After an analysis of accident sequences and listing

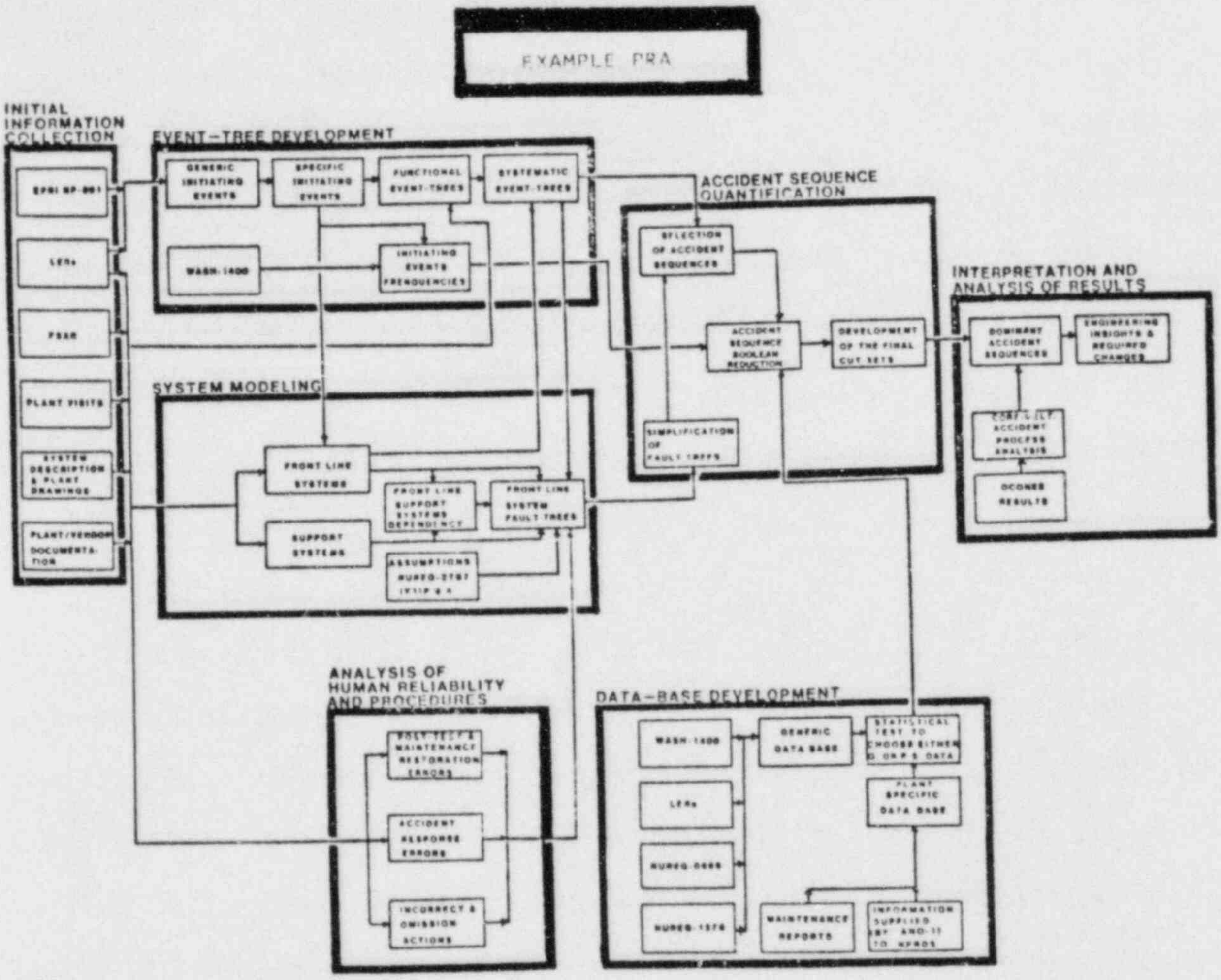

FIGURE 2. An example Flow Chart Showing One of the Practitioner Specific Methodologies.

of possible human errors, human error rates are assigned an arbitrary value, say 0.10. Then, dominant accident sequences are calculated using a critical cut-set value. Applicable human error rates are then given more realistic values and the accident sequences are recalculated to determine the extent to which human error is important to the sequence.

From the flow chart in Figure 2, an expansion can be made to see the entire structure of the PRA. The structure, showing the three degrees of specificity, is shown in Figure 3. The figures show each generic step at the top, the action steps required to perform the generic step by the PRA practitioner in the middle and finally, those inputs required to perform those action steps at its most specific, lowest degree of specificity. As can be seen, inputs can be used several times for different reasons.

Also incorporated into many of the reports were the personnel who performed the PRA itself. These persons were noted when present, and their primary function listed. However, generally, specific tasks or responsibilities within the process were not given and the PRA team and qualifications were listed as a group.

PLANS FOR FUTURE WORK

The results of the data base created in Phase 1 will be used in subsequent phases to create new methodologies to address questions that cannot currently be answered quantitatively or qualitatively by PRA. First, a list of such questions will be identified. Then, individual questions will be grouped into three major categories:

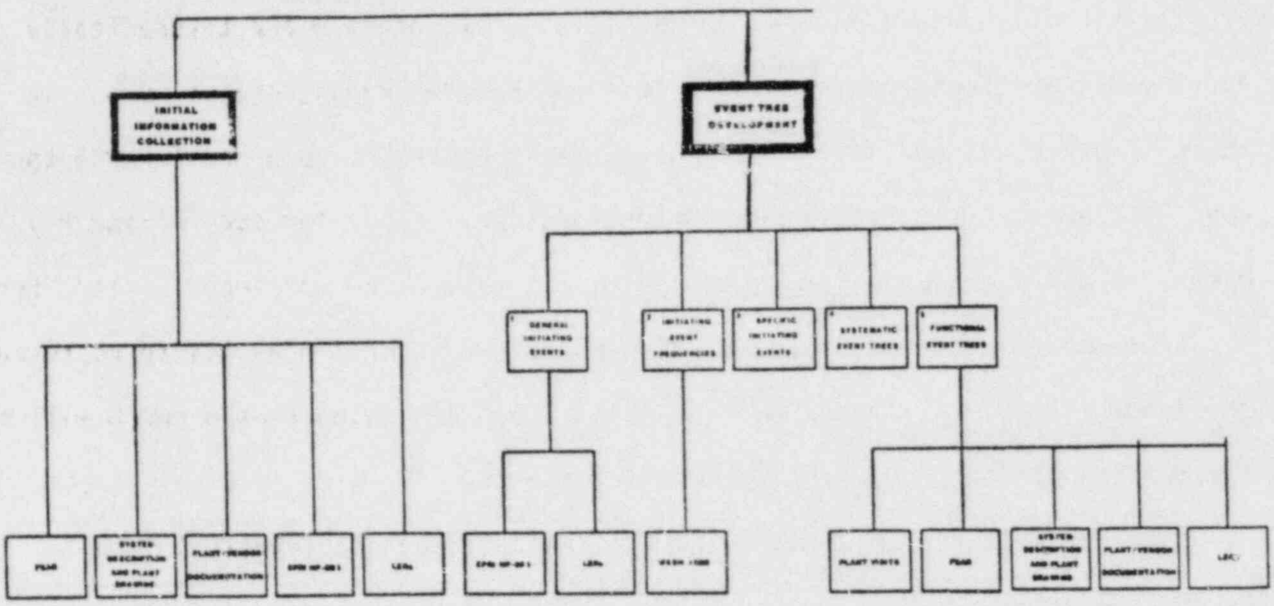

FIGURE 3. Expanded Structure of the Flow Chart Shown in Figure 2.

a. Questions which are presently addressed by PRA.

b. Questions which could be addressed with changes to structure or form of PRA.

c. Questions which cannot be addressed by PRA.

Within the three categories of questions, two of them are theoretically addressable by some form of probabilistic risk assessment; categories a and b. For category a questions, no action is necessary in order for PRA to resolve the issue. However, for category b, new methods for quantifying key questions not addressed by current PRA practices must be developed. Also, for all addressable issues, it is desirable to have a numeric quantity to represent the issue resolution. Revised PRA methodologies for categories a and b will be the subject of the second phase of this research.

The first part of the second phase of this research project will be to reorganize and categorize key regulatory issues into the three part taxonomy mentioned earlier; then, into groups which are related in terms of topical area. A group of subject matter experts familiar with human factors, human reliability analysis, and/or probabilistic risk assessment will then combine the categorized issues list with the generic structure to define appropriate entry points for revised methodologies to resolve the key issues. Finally, revised structure producing quantitative solutions, if possible, to the questions or topic areas. If it is determined that the topical area is treated in a manner which meets regulatory needs, then no changes will be made to procedures or structure. Otherwise, new specifications and structure will be developed to implement and carry out steps to sufficiently and validly answer those questions. Included in these specifications will be qualifications of analysts, needed materials and/or references, interfaces of personnel, and reporting requirements needed to complete the process.

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NUREG-75/014, WASH-1400. "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants." October 1975.

NUREG-1050. "Probabilistic Risk Assessment (PRA): Status Report and Guidance for Regulatory Application." Draft Report, February, 1984.

NUREG/CR-2728. "Interim Reliability Evaluation Program Procedures Guide." 1983.

THE USE OF HUMAN RELIABILITY DATA REPORTED IN PROBABILISTIC RISK ASSESSMENTS IN ADDRESSING HUMAN FACTORS SAFETY ISSUES*

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This paper describes a research program currently being performed by Brookhaven National Laboratory (BNL) which aims to identify and improve means of using Probabilistic Risk Assessment (PRA) results to address human factors safety issues. The long-term goal of this research (FY 1987) will be to make the development process and documentation structure of PRAs more applicable to human factors safety issues facing NRC. Of particular interest are (1) the identification of retrofit requirements, (2) development of baseline measures to evaluate them, and (3) identification of future human factors research needs.

Research has started in two phases at BNL. These steps involve (1) identifying and cataloguing the human reliability data reported in PRAs and (2) identifying and articulating human factors safety issues confronting NRC. Human factors safety issues and human reliability PRA data will be matched in order to determine how useful current PRA results are in addressing those issues. Methods of using PRA data through manipulation and combination with other data sources to address issues will also be developed. In addition, information concerning errors of commission and omission used in PRAs are being examined and reported on. In the following fiscal years, changes in the PRA process and structure proposed in related efforts will be evaluated by BNL to determine how to optimize the usefulness of PRAs as a regulatory tool. These efforts are discussed separately below.

1. Identifying and Cataloguing of PRA Human Reliability Data

The complete, published PRAs from 18 nuclear power plants were obtained through the NRC, utilities, and contractors. This resulted in acquisition of over 65 volumes of PRAs. A set of keywords referring to human reliability was developed and technical readers screened every page of each volume marking each occurrence of any keyword. Each instance of a keyword was then examined to determine if it met the criteria established for identifying human reliability data. If so, the data were entered into the computerized database described below.

1.1 Criteria for Identifying Human Reliability Data

Human reliability data are defined as:

*Work performed under the auspices of the U.S. Nuclear Regulatory Commission.

1. Point estimates of human error probabilities or rates.
2. Estimates of uncertainty bounds for human error probabilities or rates.
3. Identification and qualitative scaling of a specific performance shaping factor (PSF) (e.g., high stress) in relation to its effect on human reliability (i.e., high stress which would impair performance).

1.2 Structure of Databases

Aston-Tate's DBASE III for the IBM-PC was used to create a database to store and organize the human reliability data reported in completed PRAs. This software makes it possible to combine individual database files, count, sort, and modify data within files, and generate reports concerning specific aspects of the data within files.

One database file contains demographic information about each of the plants and its PRA. At present, this file contains information on 18 plants and their respective PRAs. These plants are:

Arkansas 1	Millstone Point 1
Big Rock Point	Oconnee 3
Browns Ferry 1	Peach Bottom 2
Calvert Cliffs 2	Seabrook
Crystal River 3	Sequoyah 1
Grand Gulf	Shoreham
Indian Point 2/3	Surry 1
Limerick	Yankee Rowe
Midland	Zion 1/2

The following demographic information is included on each plant: NRC region, year licensed, type of reactor, number of nuclear units on site, size in megawatt output (small, medium, large), architect/engineer, utility, nuclear steam supply system vendor.

The following information about the PRA is included in this database: PRA sponsor (NRC, utility, other), PRA contractor(s), years PRA was initiated and published, and scope of analysis (limited, moderate, full).

A separate database file has been created for each PRA into which human reliability data are entered. A record is created for each datum identified. The purpose of this database is to describe the way in which human reliability is considered, incorporated, and reported in PRAs. To this end, the following information about each human reliability datum reported in a PRA is provided:

1. Plant state at time of human action (normal/off-normal).
2. Impact of action on availability of safety systems (has impact/no impact).

3. Accident sequences:
   a. LOCA. (Small/medium/large) with or without failure of coolant injection/coolant recirculation.
   b. Transient. Loss of off-site power (with or without failure of coolant injection/coolant recirculation). Open primary steam relief valve, open main steam relief valve, turbine trip, main steam isolation valve closure, loss feedwater, ATWS, station blackout.
   c. Fuel cycle. Fabrication, transportation, in-plant use and storage, reprocessing, long-term disposal.
   d. External events. Seismic, fire, flood, other.

4. Personnel involved (control room operators, equipment operators, maintenance, security, instrumentation and control, chemistry, health physics, contractor, off-site response).
5. Action required (test, operate, monitor, diagnose, maintain, inspect, other).
6. Error classification (omission/commission, individual/group, knowledge-/rule-/skill-based).

1.3 Results

The importance of modeling human reliability in PRAs is acknowledged by virtually every PRA analyst team. The more recent PRAs treat human reliability as a factor in risk assessment far more extensively than in some of the earlier risk assessments. However, the PRA process is still limited in its ability to incorporate the role of human reliability in the initiation, mitigation, and contribution to accidents in nuclear power plants.

Human actions are incorporated at very low levels in fault trees. Such actions are usually described as errors of omission (e.g., operator fails to actuate HPIC system). At present, errors of commission involving inadvertent actions or cognitive errors are harder to incorporate in fault trees; thus, only a small subset of the possible errors of commission are analyzed. Regardless of the type of personnel included, operator errors are virtually the only type of human errors which are considered on a primary level, although design errors and maintenance errors are occassionally referred to.

Overall, for the kinds of human actions which are included in PRAs, little information is provided about the specific type of action or error being considered. Currently available PRAs do not classify errors in terms of commission/omission, individual/group, or knowledge-/rule-/skill-based. In most cases, the actions are described only as a failure to actuate, a failure to realign a component or system, or a testing and maintenance error.

Although PRAs comment on the importance of operator actions to risk, there appears to be little human reliability data which is actually included in the accident scenarios evaluated. There are few sources of data on human error rates and/or probabilities and the quality of this data is not high.

Thus, almost all PRA analysts report using the same generic human error estimates from common sources (e.g., NUREG/CR-1278). Many estimates are not plant-specific or even industry-specific. They are rarely reported to be modified by a scaling of PSFs during the hypothesized accident sequence although such scaling does occur.

The databases developed in this project will allow an analyst to obtain counts of human error data contained in each PRA, by accident sequence, type of action, type of error, and so on. The database has been designed to be expandable and should be a very useful tool for decision-making and regulation when documentation of human reliability estimates in PRAs is improved.

2. Identification and Articulation of Human Factors Safety Issues

As part of this effort, a comprehensive list of human factors safety issues confronting NRC was developed. This "issues list" is a major component of the overall program to improve PRAs. These issues will be used to identify data needs which can be considered in the context of increasing the applicability of PRA results to human factors regulatory actions. As such, the issues list was developed using several sources of information on NRC regulatory concerns in order to assure comprehensiveness. These sources were (1) research planning documents, (2) interviews with cognizant NRC personnel, and (3) related research efforts. Each of these is discussed below.

2.1 Research Planning Documents

A review of research planning documents was undertaken in order to develop an initial, basic issues list. The documents reviewed included "U.S. Nuclear Regulatory Commission Human Factors Program Plan," NUREG-0985; "Critical Human Factors Issues in Nuclear Power Regulation and a Recommended Comprehensive Human Factors Long-Range Plan," NUREG/CR-2833; "A Long-Term Research Plan for Human Factors Affecting Safeguards at Nuclear Power Plants," NUREG/CR-3520; and "Maintenance and Surveillance Program Plan," (Draft), May 25, 1984.

A list of human factors safety issues was developed from each document and integrated to arrive at the initial, basic issues list.

2.2 Interviews with Cognizant NRC Personnel

Assuring the comprehensiveness and usefulness of the issues list required that NRC personnel amplify and articulate issues. In order to accomplish this, NRC and BNL developed a roster of NRC personnel responsible for each area of human factors regulation and research. First, BNL attended the annual NRC Human Factors Review Group meeting in June in order to obtain a current overview of relevant NRC activities. Subsequent to the meeting, individual interviews were requested of those on the NRC personnel roster. Each interviewee received the initial, basic issues list and a set of questions aimed at amplifying and articulating the issues prior to the interviews. Over 30 NRC personnel were individually interviewed. Each interview lasted between one and two hours and usually involved candid, direct discussions of the human factors safety issues for which the interviewee was responsible. During the interview stage, many subtle needs were identified which were not explicit in the review of research planning documents.

2.3 Related Research Efforts

Other research efforts have focused on the data needs attendant upon the TMI Action Plan and NRC Generic Issues. These needs were reviewed and used to supplement the issues list in order to arrive at a final issues list.

2.4 Results

The results of this effort is a comprehensive list of current human factors safety issues which are listed in terms of a taxonomy. This list has been reviewed by NRC to assure its comprehensiveness.

3. Matching PRA Results Issues

In order to determine the applicability of current PRA results to NRC human factors safety issues, a matching scheme has been developed as is indicated in Table 1. The PRA results database is being analyzed to determine which issues can be addressed. This is done by individually classifying each issue (right column, Table 1) and matching the appropriate set of factors in the PRA results database (left column, Table 1). Initial results indicate that only a small fraction of the human factors safety issues currently confronting NRC can be addressed with current PRA results.

Table 1

PRA Results	Issues
Plant Demographics	Type of Data
PRA Information	Probabilities, Counts, etc.
Plant State	Situation or Condition
Impact on Availability	Normal, LOCA, etc.
Accident Sequence	Important Factors
Personnel Involved	Error Type, Personnel, etc.
Action Required	
Error Type	

4. Current and Future Research

In order to increase the applicability of PRA results for human factors safety issues, BNL is currently identifying approaches for applying PRA results to a broader range of issues and developing procedures for applying these results. These efforts are discussed below.

4.1 Identification of Approaches for Applying PRA Results

Evaluation of plant specific and generic retrofit requirements at power plants is a prime interest of NRC. BNL is developing approaches to (1) identifying retrofit requirements, (2) providing baseline measures useful in evaluating retrofits, and (3) identifying immediate and future human factors research needs. This is being done by continuing to group and analyze issues in such a way as to resolve discrepancies between existing PRA results and data needed to resolve human factors safety issues. This analysis will include consideration of (1) directly applicable PRA results, (2) PRA results which can be indirectly applied by manipulation of reported data, and (3) PRA results which can be useful when supplemented by human factors data from other sources.

4.2 Development of Techniques for Implementing the Approaches Identified

Issues which were not found to be addressed by current PRA results, but which would be addressed through revisions in the current PRA development process and documentation structure, will be considered. BNL will develop approproate techniques, models, and/or procedures for applying the PRA results attendant upon those revisions to a broader range of human factors safety issues. It is anticipated that each technique will specify input data, data processing protocols, and criteria for addressing particular issues of interest. These approaches will be field tested during FY 1986.

5. Conclusions

This project will arrive at three useful goals. First, the database developed by BNL to include all results reported in PRAs will be useful to regulators, system designers, utilities, and PRA analysts in identifying and articulating PRA human factors data. It is designed to be expandable and will accommodate a vast range of human factors data expected to be useful. Second, the issues list will serve as a reference to NRC personnel in discussing human factors safety issues and related research. Third, the process and structure of PRAs will be improved to make these documents more useful than is currently the case. These goals will make future PRAs useful for a broad range of applications which are not currently available and assist NRC in resolving human factors safety issues.

VITAL EQUIPMENT DETERMINATION TECHNIQUES RESEARCH STUDY

by

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ABSTRACT

This paper presents the results to date of the Vital Equipment Determination Techniques Research Study (VEDTRS) performed at the Los Alamos National Laboratory. This study is associated closely with the Vital Area Analysis Program that Los Alamos has been conducting for the Nuclear Regulatory Commission (NRC) since 1978 to identify the vital areas at nuclear power plants using fault-tree analysis. The VEDTRS started in 1982 with a literature survey of the available information about 11 major sabotage fault-tree assumptions that were identified for reexamination by an NRC interoffice working group. The FY 1983--FY 1984 effort was to continue the literature survey, compare the current fault trees with identified information, and prioritize the topics for research. Los Alamos also performed well-focused internal research needed to resolve topics according to their priority.

I. INTRODUCTION

In 1978, the Nuclear Regulatory Commission (NRC) requested technical assistance from the Los Alamos National Laboratory in determining vital areas based on 10 CFR 100 release criteria for all commercial nuclear power plants in the United States. A systematic analysis based on a fault-tree methodology was developed at Sandia National Laboratories, Albuquerque (SNLA), in the early 1970's for the NRC Office of Research (RES). Since 1978, Los Alamos has applied this analysis method to specific plants for the Office of Nuclear Reactor Regulation (NRR) and, more recently, for the Office of Nuclear Material Safety and Safeguards (NMSS). This technique has proven to be an excellent tool for performing detailed and systematic analyses of complex nuclear power plants.

The fault-tree methodology uses the Set Equation Transformation System (SETS) computer code[1] to determine cut sets and provide usable information on vital area determination. The construction of the fault tree is central to the entire program, and accurate representation of the plant is essential for credible results.

Vital area analyses (VAAs) have been performed on all operating nuclear plants and on several plants nearing completion of construction. The thrust of current VAA work is to provide a consistent and accurate analysis of all nuclear power plants. Because NRC's direction has changed with respect to VAAs, some analyses performed before 1981 are being redone to provide information that was not required when the original analyses were completed. In addition, plants receiving operating licenses are analyzed as they come on line. Consistent analyses of all plants will be available within a few years and are being used by NMSS in its Regulatory Effectiveness Review (RER) program.

The Vital Equipment Determination Techniques Research Study (VEDTRS) began in 1982 as a result of the recommendations of the Vital Area Determination Working Group,[2] which was convened to examine the vital area techniques used by Los Alamos. The objectives of the VEDTRS work are threefold.

(1) Identify, to the extent possible, all technical limitations of the current vital area identification process.
(2) Develop approaches to eliminate or minimize the effect of those limitations on the results of the process. If necessary research and engineering analysis has to be completed, it may be possible to do this by incorporating these results directly. Otherwise, appropriate assumptions may have to be made.
(3) Identify research or analysis programs that may serve to refine the assumptions and approaches developed to eliminate or minimize the effect of technical limitations.

II. PROGRAM SCOPE

Sabotage analysis is complicated by an inability to eliminate possible sequences of events based on single failure criteria or their low probability of occurrence. Therefore, in order to make the analysis manageable, some simplifying assumptions were necessary. Also, new information that affects the decisions made in constructing the fault trees eventually will become available. This necessitates a reexamination of the fault-tree assumptions and can change the vital area solutions based on these new findings.

The assumptions to be examined were identified by an independant NRC working group formed for the review of the VAA. The working group identified 11 topics for reexamination. These topics are listed in Table I. The initial effort of the vital area determination research was to survey existing literature for information related to these topics and to document the information. Ongoing research projects promising information of interest also were identified.

The 11 topics were reviewed every fiscal year and placed into three groups based on the results of the research.

(A) Topics for which sufficient information already exists to resolve any uncertainties.

(B) Topics for which ongoing research appears adequate for resolution of uncertainties.

(C) Topics for which there is inadequate information or ongoing research to address uncertainties from the VAA perspective.

An up-to-date categorization of the topics is given in Table II. Some topics are very broad in scope and include several subtopics (for example, use of best-estimate analyses.) In these cases, different subtopics fall into different categories.

The next step in the analysis procedure was to set priorities for research topics. These priorities were based on the topic's effect on the fault trees and ease of resolution. The topics fell into three groupings.

(1) Topics resolvable with limited effort and affecting the fault tree significantly.

(2) Topics requiring considerable work and having significant effect.

(3) Topics that are not considered to have major effect because they do not represent direct, high-probability sabotage success paths.

The priority grouping is shown in Table III and will guide future research on this project.

TABLE I

ELEVEN TOPICS FOR THE VITAL EQUIPMENT DETERMINATION
TECHNIQUES RESEARCH STUDY

1. Identifying individual safety-related cables in cable trays.

2. Disabling complete cable trays.

3. Disabling systems needed during shutdown or refueling conditions.

4. Disabling sensor systems, instrumentation, and nonsafety-related control systems.

5. Treating spatially extended systems and components [that is, piping; electrical distribution; and heating, ventilating, and air conditioning (HVAC) systems].

6. Scenarios involving air systems.

7. Disabling electrical equipment by grounding or lifting of grounds.

8. Relating best-estimate analyses of plant responses to system failures to the corresponding Final Safety Analysis Report (FSAR) analyses.

9. The effective inclusion of random events, such as anticipated transients, in fault-tree methodologies.

10. Possible system failures after which stable hot shutdown cannot be maintained indefinitely.

11. Considering the use of nonsafety-related equipment, unanalyzed procedures, or operator ingenuity to recover from system failures.

TABLE II

CATEGORIZATION OF RESEARCH TOPICS[a]

Research Topic	Category
1. Individual cables	A
2. Cable trays	B
3. Shutdown or refueling systems	A, B
4. Sensor systems, instrumentation, nonsafety-related control systems	B, C
5. Spatially extended systems	C
6. Air system	C
7. Electrical grounding or lifting of grounds	C
8. Use of best-estimate analyses	A, B, C
9. Random events	A, B
10. Stable hot shutdown requirements	B, C
11. Use of nonsafety-related equipment, unanalyzed procedures, or operator ingenuity	B, C

[a]The categories are as follows.
(A) Topics for which sufficient information already exists to resolve any uncertainties.
(B) Topics for which ongoing research appears adequate for resolution of uncertainties.
(C) Topics for which there is inadequate information or ongoing research to address uncertainties from the VAA perspective.

III. RESULTS TO DATE

To date, only research topic 1 (Identification of Safety-Related Cables in Cable Trays) was well resolved. This topic validated the current assumption that safety-related cables cannot be identified in cable trays. Therefore, we found no effect on the fault-tree modeling. Significant findings for other research topics are described below.

TABLE III

RECOMMENDED PRIORITY OF RESEARCH TOPICS

Group 1 - Potential major effect, reasonable sabotage paths, limited research effort.

Research Topic 3 - Disabling systems needed during shutdown or refueling conditions

2 - Disabling complete cable trays

9 - The effective inclusion of random events, such as anticipated transients, in fault-tree methodologies

10 - Possible system failures after which stable hot shutdown cannot be maintained indefinitely

Group 2 - Potential major effect, reasonable sabotage paths, more research effort.

Research Topic 4 - Disabling sensor systems, instrumentation, and nonsafety-related control systems

8 - Relating best-estimate analysis of plant responses to system failures to the corresponding FSAR analysis

11 - Considering the use of nonsafety-related equipment, unanalyzed procedures, or operator ingenuity to recover from system failures

Group 3 - Potential effect, least likely paths, more research effort.

Research Topic 6 - Scenarios involving air systems

7 - Disabling electrical equipment by grounding or lifting of grounds

5 - Treating spatially extended systems and components

A. Research Topic 3. Disabling Systems Needed During Shutdown or Refueling Conditions

Current fault-tree analysis assumes that a reactor operating at full power is in the most vulnerable configuration. Therefore, vital areas identified based on this assumption would include as a subset vital areas derived from all other reactor operating modes. Analysis of reactor plants in cold shutdown mode (0% rated power, reactor at subcritical, and average coolant temperature less than 200°F) has been undertaken, and new sabotage scenarios have been identified. Three nuclear power plants, a boiling water reactor (BWR) and two pressurized water reactors (PWRs), were analyzed by Los Alamos and are discussed briefly.

For the BWR/3 Mark I plant, it was assumed that the decay heat levels and amount of coolant loss in a shutdown loss-of-coolant accident (LOCA) are such that use of the control rod drive (CRD) pumps for makeup would not prevent core damage. The location solution of this fault tree yielded an area for which vital designation is appropriate. This new area appeared in all protection sets which were generated by taking complement of the location solution. One new solution of the facility system equation involves protecting the integrity of at least one loop of RHR or one loop of core spray. Another solution would be to protect both RHR loops to prevent a LOCA. Either solution assures the integrity of the RCS boundary or the integrity of the LOCA mitigating system.

For the Westinghouse and Babcock and Wilcox (B&W) PWR plants, two assumptions were made in preparing the fault tree.

- The residual heat removal (RHR) pump used to cause the loss of reactor coolant is disabled in the "on" position at the appropriate 6900-V switchgear cabinet.
- The disabled RHR loop is damaged to the extent that the operators cannot realign it for safety injection.

The location solution indicated that there was no new Type I area (where access to this area alone would be needed to cause 10 CFR 100 release). The protection strategies involve the integrity of at least one RHR loop or one charging pump loop in both plants. In addition, one can also choose to protect both RHR loops to prevent a LOCA.

It should be noted that the vital area solutions from these analyses are very plant specific. Los Alamos intends to resolve this topic by analyzing other typical plants, including a BWR/6 Mark II plant and a PWR designed by Combustion Engineering (CE). Our recommendation at this time is to continue current fault-tree practices until a generic conclusion can be made from more cold shutdown analyses.

B. Research Topic 8. Best Estimate Analysis

Current fault-tree analysis assumes that a release without core melt would not cause doses in excess of 10 CFR 100 limits. This release event was included in the generic sabotage fault trees previously developed by the SNLA and Science Applications, Inc. The release without core melt could result from either one of the two accident scenarios: a small LOCA outside the containment or a LOCA inside the containment with subsequent containment failure. The former case could be a reactor coolant system interfacing line break with associated isolation valves disabled open. The interfacing line could be the letdown line for a PWR or the reactor water cleanup line for a BWR. The latter case could be a power-operated relief valve LOCA in a PWR or an automatic depressurization system valve LOCA in a BWR with subsequent loss of containment. In both scenarios, the core is assumed to remain covered for an extended period with continuous reactor coolant makeup so that no fuel is damaged.

The analysis considered the radionuclide inventory, the iodine spike phenomenon, atmospheric dispersion factors, and radionuclide containment capabilities. The results show that the release from an instantaneous total loss of the coolant inventory would not cause 10 CFR 100 doses for two typical nuclear power plants. The results also indicate that doses from the PWR differ from those from the BWR by approximately a factor of 5. In any case, the maximum resulting doses are less than the 10 CFR 100 limiting doses by a factor of 10 to 100. Thus, the analysis validated the current assumption and practice for the release without core melt event.

In the current VAA, the containment is assumed to be breached if core melt occurs. However, the question of containment integrity with no core melt is of some concern. If a PWR containment is pressurized by a LOCA, conservative assumptions on containment pressure rise require containment cooling systems to prevent possible containment breach. If the containment is breached, the depressurization will cause RHR pumps to cavitate in the recirculation mode, threatening a loss of long-term cooling. Thus, containment cooling sysems appear in some PWR fault trees.

Past research has indicated that containment strengths are far in excess of FSAR values, and current work in the severe accident area should provide information on penetration strength. A recommendation on containment modeling will be made when this information becomes available. It should be noted that no change from a Type II (where access to a combination of these areas would be needed to cause a 10 CFR 100 release) to a Type I area at any plant is expected. The possible results of this work would be changes in Type II areas at a few plants. The current fault trees appear to be conservative and provide adequate vital area determination.

C. Research Topic 9. The Effective Inclusion of Random Events

Current fault trees do not include the occurrence of random equipment failures or violent natural phenomena concurrent with an adversary attempt because it is assumed that a person cannot depend on good luck to achieve his objectives. This assumption bounds the analysis by permitting only consideration of technical specification requirements for plant operation with minimum equipment.

Certain classes of random events (within technical specification limits), such as anticipated transients and equipment unavailability because of maintenance, are being examined in the study. SNLA has assessed the effect of maintenance and testing (M&T) on vital areas at a BWR and a PWR. We reviewed this study, and the location solution for the PWR plant yielded new Type I areas where at least one of the redundant motor-driven auxiliary feedwater trains has to be protected while the turbine-driven auxiliary feedwater train is under maintenance. Other M&T activities resulted in changes to the location combinations for both plants. As a result of possible term elimination in Boolean algebra in the fault-tree analyses, Los Alamos identified at least two specific plant configurations for a PWR where a new vital area might be introduced because of the unavailability of one train of the auxiliary feedwater system. Los Alamos already has found a new vital area in a PWR that has one of these two specific configurations. The recommendation at this time is to assure protection strategies for the auxiliary feedwater systems for all PWRs.

D. Research Topic 10. Possible System Failures After Which Stable Hot Shutdown Cannot Be Maintained Indefinitely

The current NRC staff position on VAA considers reaching stable hot standby as adequate for preventing core damage. However, hot shutdown may not be maintained indefinitely after certain events, and some consideration should be given to the plant's capability to achieve a stable cold shutdown.

One area of concern in maintaining stable hot standby conditions is loss of reactor coolant system (RCS) pump seals during a loss of offsite power. Currently, Los Alamos, under the NRC's direction, does not include seal injection water and component cooling water to the pump during hot standby in the vital area fault trees. If the RCS pump seal failure were induced by a loss of all RCS pump seal injection and component cooling water, the RCS inventory integrity would be lost. This would require that the charging water, onsite ac power sources, and associated essential auxiliaries be protected.

Los Alamos[3] simulated the primary system response for a Westinghouse PWR plant, assuming that RCS pump seal failure occurs 30 min after loss of cooling water with an initial leak rate of 1200 gal/min. The results from the analysis using TRAC/PF1 show that the seal leakage would cause core uncovery at about 3 hrs after the seal failure. Los Alamos has maintained contact with the task lead office at the NRR on Generic Safety Issue (GSI)-23, "Reactor Coolant Pump Seal Failures," so that findings from this task can be used to refine assumptions for seal failure modes and leak rates. The current fault tree would require modification for at least Westinghouse PWR fault trees if RCS pump seal research indicates a high likelihood of leakage on loss of cooling and injection water. Any decision should await the results of current work at the Brookhaven National Laboratory on GSI-23 and the pump analysis at Westinghouse.

Item II.E.3.1 in the "Clarification of TMI Action Plan Requirements" (NRC report NUREG-0737)[4] is a statement of clarification for the emergency power supply for pressurizer heaters. In NUREG-0737 the requirements and bases are stated as follows: "provide the capability to supply from either offsite or emergency power a sufficient number of pressurizer heaters and their controls to establish and maintain natural circulation at a hot standby condition."[4] Current fault trees do not consider the need for pressurizer heaters and sprays as a mitigating system for transients combined with loss of offsite power. To study the need for these components for a stable hot standby condition, we plan to perform a literature search, or we will perform best-estimate analyses if needed.

Current fault trees do not include the primary coolant charging system as a mitigating system for transients coupled with loss of offsite power. Los Alamos plans to investigate the ability of operators to maintain the reactor at a hot standby condition without charging water. We make no recommendation concerning changes at this time.

IV. FUTURE WORK

Future effort in this program will include work on the more extensive research projects and following ongoing programs that seem to offer guidance on VAA topics. Some, or considerable extension of some, ongoing research may be required to address specific VAA concerns. Thus, the overall outlook for this project is a gradual resolution of outstanding VAA questions through a combination of monitoring and adapting outside research and performing some well-focused internal work to extend and augment other efforts.

Next fiscal year, Los Alamos will perform the research highlighted in Table IV. We also will study the validity of other assumptions not covered in the 11 research topics. Los Alamos will determine the scope and effect of these new topics on the VAA and will make recommendations on prioritizing these new topics in research.

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TABLE IV

LOS ALAMOS FY 1984--FY 1985 RESEARCH ACTIVITIES

Research Topics	Activities
Shutdown LOCA (Topic 3)	• Verify current sabotage scenarios • Perform additional LOCA analyses for one CE PWR plant and one BWR/6 plant • Use TRAC to analyze RHR LOCA
Cable Tray Destruction (Topic 2)	• Analyze Appendix R licensee documentation
Random Events (Topic 9)	• Effect on plant of maintenance and testing of auxiliary feed-water system.
Stable Hot Standby (Topic 10)	• Use TRAC to analyze --Charging water requirement --Pressurizer heater and spray requirement --RCP seal leak transient

"Occupational Dose Reduction Developments and Data Collected at Nuclear Power Plants"

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Introduction

Brookhaven National Laboratory has been contracted by the Occupational Radiation Protection Branch of the Nuclear Regulatory Commission to study occupational dose reduction at nuclear power plants. This project is entitled "Technical Assistance: Occupational Dose Reduction at Nuclear Power Plants". Its purpose is to provide information to industry which will be useful in preplanning for radiation protection during maintenance, operations and inspections. The objectives of this project are to:

- Identify repetitive high dose jobs, their collective dose range, and their respective dose reduction techniques,
- Investigate the use of low maintenance and high reliability equipment,
- Recommend improved radioactive waste handling equipment and procedures,
- Examine current ALARA incentives and recommend new positive steps to provide additional dose reduction incentives,
- Compile a NUREG Report, and
- Compile an ALARA Handbook.

The NUREG Report will summarize our findings on the above objectives. The ALARA Handbook, which will be utilized mainly by utility Health Physicist and ALARA Coordinators, will be a loose leaf type handbook which will contain:

- Data and techniques for high exposure jobs,
- Cost-benefit calculations for dose reduction modifications,
- ALARA procedures,
- Listing of ALARA equipment,
- Case histories of innovative ALARA techniques,
- A glossary of dose reduction references, and
- A list of persons interested in ALARA at power reactors

Prior to publication of these reports, the information will be reviewed by the power plant personnel whom we interviewed, our BNL-Industry Dose Reduction Advisory Committee and the NRC. The Dose Reduction Advisory Committee is made up of a representative from AIF, EEI, EPRI, INPO, Bechtel, G.E., Westinghouse, Commonwealth Edison, Northeast Utilities and T.V.A. The information presented here is preliminary and has not yet been reviewed; therefore, it may be subject to change in our final report. The NUREG report should be available in the beginning of 1985 and the ALARA Handbook in late 1985.

Plants Visited

Ten nuclear sites were visited by two Brookhaven health physicists with past nuclear power plant experience to collect the needed information. This encompassed 19 nuclear units, which were selected based on the availability of several years of computerized job-specific data on occupational doses.

Table 1 shows the plants visited. This table includes the Power Rating to indicate size, the Years of Operation to indicate age and the Plant-Years of Data to indicate the weight of that plant's collective dose data. The six Westinghouse units had 28 plant-years of man-rem data, the six General Electric units had 12 plant-years, the three Combustion Engineering units had 15 plant-years and the three Babcock and Wilcox units had 9 plant-years.

Table 1
Nuclear Plants Visited

Westinghouse Plants

Plant Name	Power Rating (MWe)	Years of Operation	Plant-Years of Data
Zion 1 & 2	1040	10	10
Turkey Point 3 & 4	666	11	3
Haddam Neck	600	17	7
Kewaunee	535	10	8
6 Units	-	-	28 Total

General Electric Plants

Quad Cities 1 & 2	789	12	6
Milstone 1	660	14	2
Browns Ferry 1,2 & 3	1076	9	4
6 Units	-	-	12 Total

Combustion Engineering Plants

St. Lucie 1	777	7	6
Milstone 1	870	8	4
Maine Yankee 1	825	11	5
3 Units	-	-	15 Total

Babcock and Wilcox Plants

Oconee 1, 2 & 3	860	10	9
3 Units	-	-	9 Total

High-Dose Jobs and Dose Reduction Techniques

The first objective of this project has been the identification of repetitive high-dose jobs, their collective dose equivalent range, and dose reduction techniques applicable to each job. This information will enable industry and the NRC to focus on these major dose-reduction targets. Fifty high-dose jobs were identified and studied.

To reduce the total station collective dose equivalent, the logical approach is to identify the repetitive jobs which cause the highest collective dose equivalent, determine their relative dose-reduction potential and then implement the most cost beneficial dose-reduction techniques. Therefore, a list of the high-dose jobs, their relative dose-reduction potential and the associated dose-reduction techniques to be evaluated is important to reducing total station dose equivalent.

The collective dose equivalent for repetitive high-dose jobs were obtained by preparing a generic description of each job prior to the plant visits. These descriptions of the high-dose jobs were modified by station personnel to better define the work associated with these doses. The station ALARA coordinator retrieved computer printouts, ALARA reports and/or letters which contained the needed data on high dose jobs. The Collective Dose Summaries for High Dose Jobs for General Electric and Westinghouse plants are shown in Tables 2 & 3. The dose range per job is indicative of the dose-reduction potential for that job.

Table 2
General Electric Boiling Water Reactors
Collective Dose Summaries for High Dose Jobs

Job Title	Integrated Dose (Man-Rem) Minimum	Maximum	Average	Population Size
Reactor Assembly/Disassembly	7.8	51.0	18.6	11
Fuel Shuffle/Sipping & Inspection	3.8	58.4	17.3	11
CRD Removal/Rebuild & Replacement	6.3	229.0	74.2	11
Recirculation Pump Seal Repair	1.5	22.7	8.1	11
*Torus Repair Inspection and Modification	125.1	597.9	313.0	11
*Reactor Water Cleanup System Repair	9.3	195.6	83.2	9
Turbine Overhaul	0.5	14.6	6.0	8

* Dose Per Cycle = Outage + Routine Operations

Table 3

Westinghouse Pressurized Water Reactors
Collective Dose Summaries For High Dose Jobs

Job Title	Integrated Dose (Man-Rem) Minimum	Maximum	Average	Population Size
Reactor Assembly/Disassembly	12.2	120.6	50.1	19
Fuel Shuffle & Inspections	3.6	15.5	9.4	12
Steam Generator Manway Removal/Replacement	5.7	51.1	18.3	14
Eddy Current Testing (Steam Generators)	5.9	117.6	42.1	16
Reactor Coolant Pump Seal Repair	4.3	31.3	16.6	13
Secondary Steam Generator Inspection & Repair	2.3	40.8	12.3	15
*Chemical Volume & Control System Repair	0.8	22.2	10.5	16

* Dose Per Cycle = Outage + Routine Operation

The dose reduction techniques for repetitive high-dose jobs were obtained by questioning maintenance, engineering and the health physics personnel on the "tricks of the trade" to reduce exposures and the spread of contamination. The listing of the consolidated dose-reduction techniques can be used in preplanning for radiation protection during these activities. The Dose Reduction Data Sheet for PWR-Reactor Coolant Pump Seal Repair(3) is given below as an example of the dose-reduction data sheets we are developing.

PWR REPETITIVE HIGH DOSE JOB

DOSE REDUCTION DATA SHEET

JOB TITLE: Reactor Coolant Pump Seal Replacement

JOB DESCRIPTION: Outage or forced outage reactor coolant pump seal replacement. Includes: auxiliary piping and coupling removal; oil pan removal; coupling, runners, seals and seal housing or seal package removal; seal area cleaning, seal package replacement; heat fit coupling; concentricity alignment; oil pan replacement; replace auxiliary piping and replace oil. Excludes: exposures associated with vibration measurements, pump ISI inspections, pump modifications (e.g. fire protection oil drip pans), reinsulation, painting, and motor inspections and repair.

COLLECTIVE DOSE:

REACTOR SUPPLIER	MINIMUM MAN-REM	MAXIMUM MAN-REM	AVERAGE MAN-REM
Westinghouse	4.3	31.3	16.6

DOSE REDUCTION TECHNIQUES (Dose Rate Reduction):

- Steam generator in wet layup
- Evaluate shielding of local "hot spots"
- Lead blankets on grating over "hot" pipes

DOSE REDUCTION TECHNIQUES (Timesaving):

- Dedicated RCP tool boxes
- RCP seal replacement video tape
- Temporary deck between grating and flange gap
- Pneumatic torque wrench for flange
- Four ultra-small tracked chainfalls to replace seal lift rig

DOSE REDUCTION TECHNIQUES (Contamination Reduction):

- Periodically mop plastic covered grating
- Hang plastic sheet walls from rails and erect walls around contaminated parts storage area
- Portable doghouse enclosure with vacuum cleaner for ventilation to clean small parts
- Large contaminated parts cleaned over blotter paper in parts storage area
- Restrict access to area

High Reliability and Low Maintenance Equipment

The second objective has been to investigate the use of equipment reliability data, including dose received in component repair, and to determine if this data is used by maintenance and engineering personnel for purposes of dose reduction during equipment selection.

Knowing which components contribute to high maintenance or repair dose will make it possible to evaluate the value of reliability improvements to dose-reduction actions. Since routine and non-routine maintenance activities at nuclear power plants contribute about 70 to 80% of the total station exposure, reducing the amount of maintenance and repair via the use of higher reliability and low maintenance equipment may be important.

In order to accomplish this objective we questioned engineers to determine to what extent the repair dose is considered in making component selection. Also feedback between maintenance and engineering on component reliability was investigated. Lastly, the extent to which station personnel have modified or replaced components which had high maintenance or repair dose was investigated. This was accomplished by questioning the station health physics, maintenance and engineering personnel and architect engineers on the:

- Availability of nuclear plant reliability data (NPRD),
- Availability of component repair dose data,
- Application of NPRD and repair dose data to equipment replacement and selection,
- Application of NPRD and repair dose data to preventative maintenance programs,
- Methodology used to identify unreliable equipment, and
- Nature of the feedback loop on equipment reliability from maintenance worker to architect-engineers.

In regards to whether repair and maintenance dose is considered during the component selection we found that it is of secondary concern. Equipment selection is somewhat subjective. The major considerations are factors such as: cost, availability, qualification to required specifications, past experiences and reputation of manufacturer, and use of equipment similar to existing equipment for purposes of inventory and training consolidation. The use of reliable or low maintenance equipment is a general policy for selection of nuclear grade equipment. However, high reliability is stressed for purposes of plant availability. Related reduction in repair dose is a secondary benefit along with labor savings. These benefits are rarely quantified for purposes of equipment selection.

In regards to the transfer of component failure information and repair dose data from maintenance and health physics respectively, to engineering, and the transfer of this information from the utility engineers to the nuclear steam system suppliers (NSSS) and architect-engineers (A/Es), we found that this information flows informally within the utility but primarily gets to the A/E-NSSS if they obtain it from plant visits or from informal conversation.

The component repair dose data is available from some health physics computerized job dose tracking programs and is verbally transmitted to maintenance and engineering staff if requested. A few utilities have published selected component dose data in ALARA reports. However, the volume of individual component repair data and the lack of corresponding details of the repair preclude the use of this data by engineers.

The equipment reliability data is available in various forms. Equipment reliability data can be found in: licensee event report (LER) summaries published by the NRC; NPRD summaries published by INPO; computerized listings of equipment work requests being developed by utilities; and equipment failure data bases maintained by NSSS and A/E firms. However, specific component data in the appropriate format is difficult to retrieve. Therefore this data is not being widely used by engineers.

Lastly, unreliable components are being modified and replaced. In general, good communication exists between station maintenance and engineering personnel. In addition the NRC and NSSS send out bulletins on generic failure problems. Examples of unreliable components which were modified or replaced at the stations visited are: pump seals e.g. reactor coolant pumps, recirculation pumps, RWCU pumps, and charging pumps; valves which had repetitive leakage; pressure and level transmitters; and fuel transfer equipment reliability modifications.

Radwaste Handling Improvements

The third objective has been the identification of possible improvements in radioactive waste handling equipment and procedures which could reduce collective dose equivalent. Numerous radwaste packaging improvements were examined.

Radwaste operations contribute about 5-10% of the total station dose. This does not represent a major contribution to the total station dose. However, radwaste handlers and radwaste operators are a critical workgroup in that many approach their administrative dose limits and must be limited from further radiation work. Therefore, radwaste dose-reduction improvements are needed.

The major radwaste improvements were investigated by preparing a pre-selected list of equipment and procedural dose-reduction improvements. The types of waste investigated were bead and powered resins, evaporator bottoms, tank and filter sludge, spent filters/cartridges and dry active waste. Our investigation was restricted to the packaging, prepare-ready-store and truck loading stages of radwaste processing. The radwaste supervisor and ALARA coordinator reviewed the list of dose-reduction improvements and indicated those utilized and whether they considered them successful from a dose reduction standpoint. Table 4 indicates the success rate and the number of plants utilizing the radwaste dose-reduction improvements.

Table 4
Dose-Reduction Improvements

Radwaste Handling

Radwaste Improvement	Success/No. Plants Utilized*
1. Radwaste Handlers	5/5
2. Management Policy and Program	5/5
3. Lead Glass or Water Windows	4/4
4. Shielded Fork Truck	4/4
5. Remote Drum Decontamination	2/2
6. Radwaste Foreman	8/9
7. New Compactor	7/8
8. Shielded Storage Bays and Doors	6/7
9. Remote Visual Monitoring	6/7
10. Mobile Solidification System	4/5
11. Shielded Drum or Transfer Cask	7/9
12. Storage Segregation by Radiation & Type	6/8
13. Radwaste Engineer	5/7
14. Remote Level, Radiation and Contamination Monitoring	5/7
15. Remote Mixing and Capping Stations	4/6
16. Optimized Use of Filters & Resins	3/5
17. Trash Sorting Area	3/5

* Ten plants surveyed.

ALARA Incentives

The fourth objective has been the examination of current ALARA incentives and the recommendation of new steps to provide additional incentives for dose reduction. This includes identification of the important ALARA incentives, their impact on the plant ALARA programs and what can be done by the NRC and industry to improve them. The relative importance of the ALARA incentives for plant managers and workers were evaluated. In addition, the relative importance of the key components of an ALARA program were examined.

As health physicists we have strong incentives for reducing exposures since this is so basic to our profession. However, the operators of an electric generating plant have very powerful monetary incentives, which at times are in competition with dose-reduction objectives. Since you must have management support to have an effective ALARA program, a determination of managers' ALARA incentives was considered important.

A preselected listing of managers' ALARA incentives was prepared and these incentives were rated by ten plant managers, ten maintenance supervisors and ten radiation protection managers. Table 5 indicates the relative priority and the number of plants which utilized these incentives.

Table 5
Manager's ALARA Incentives

Manager's ALARA Incentives	Priority[a]			No. Plants Utilized[b]
	High	Medium	Low	
1. Increased Usage of Experienced Workers	26	2	2	9
2. Improved Personnel Relations Due to Management's Concern for Health & Safety	25	4	1	10
3. Beneficial Performance Review for Meeting Performance Goal in Dose Reduction	22	7	1	9
4. Monetary Savings from Critical Path & Labor Savings	23	1	6	8
5. Humanitarian Considerations	21	6	3	10
6. Decreased Usage of Contractors	21	2	7	6

Table 5 cont'd

Manager's ALARA Incentives

Manager's ALARA Incentives	Priority[a] High	Medium	Low	No. Plants Utilized[b]
7. Avoid Inspection Findings for Not Complying with FSAR ALARA Requirements	18	9	3	8
8. Avoid Probable Causation Liability Suits	15	12	3	5
9. National Reputation for Low Plant Doses	16	7	7	7
10. Good Public Relations	14	8	8	7
11. Recognition for Receiving INPO's Good Practice in ALARA	11	7	12	6

[a] Thirty plant personnel rated the priority of the incentives.
[b] Ten plants visited

ALARA is everyones responsibility, and the more minds and hands which are working towards its cause the greater will be a station's dose reduction. Therefore the importance of various worker ALARA awareness techniques was determined.

Again a preselected listing of workers' ALARA awareness techniques was prepared and these techniques were rated by ten plant managers, ten maintenance supervisors and ten radiation protection managers. Table 6 indicates the relative priority and the number of plants which utilized the worker ALARA awareness techniques.

Table 6
Worker ALARA Awareness Techniques

Worker's ALARA Awareness Techniques	Priority High	Medium	Low	No. Plants Utilized[b]
1. Worker Involvement in ALARA Job Reviews	24	5	1	9
2. Visible ALARA Coordinator	23	3	4	9
3. Publicizing ALARA Suggestion Implementation	19	8	3	7

Table 6 cont'd
Worker ALARA Awareness Techniques

Worker's ALARA Awareness Techniques	Priority[a] High	Medium	Low	No. Plants Utilized[b]
4. Worker ALARA Suggestion Program and Awards	18	8	4	7
5. Publicize Workers Exposure and Plant Dose vs. Annual Goal	14	9	7	6
6. Visible ALARA Office	7	8	15	3
7. ALARA Posters	6	15	9	7
8. ALARA T-shirts, Hats & Pens	7	4	19	3

[a] Thirty plant personnel rated the priority of the techniques.
[b] Ten plants visited

Lastly, the key components of an ALARA program were examined. A listing of the key components was prepared from NUREG CR-3254 entitled "Licensee Programs for Maintaining Occupational Radiation ALARA"(1) and a paper given on the topic at the 1982 Westinghouse REM Seminar.(2) Each component was rated by ten radiation protection managers and ten ALARA coordinators. Table 7 indicates the relative priority and the number of plants which utilized the ALARA program key components.

Table 7
ALARA Program Key Components

ALARA Program Component	Priority[a] High	Medium	Low	No. Plants Utilized[b]
1. ALARA Policy and Management Commitment	19	0	1	10
2. ALARA Data Base System	19	0	1	10
3. ALARA Job Review	18	2	0	10
4. ALARA Design Reviews	18	2	0	8
5. ALARA Coordinator	18	0	2	9
6. Goals and Tracking Systems	17	3	0	10
7. H.P. Technician ALARA training	17	3	0	7
8. Craft Job Specific ALARA Training	17	1	2	8
9. Engineer ALARA Training	17	1	2	4

Table 7 cont'd

ALARA Program Key Components

ALARA Program Component	Priority[a]			No. Plants Utilized[b]
	High	Medium	Low	
10. Annual or Outage ALARA Report	14	6	0	10
11. General Employee ALARA Training	14	6	0	8
12. ALARA Committee	14	3	3	9
13. ALARA Suggestion Program	14	4	2	7
14. ALARA Organization & Responsibilities	12	4	3	8
15. ALARA Program Evaluation & Audit	11	7	2	7
16. Job Specific ALARA Procedures	11	4	5	3
17. Administrator ALARA Training	9	5	6	5
18. Cost/Benefit Methodology for Man-rem Savings	8	7	5	5

[a] Twenty plant personnel rated priority of components.
[b] Ten plants surveyed.

Summary

Occupational dose reduction developments and data collected at nuclear power plants have been described. Written descriptions of repetitive high dose jobs, their collective dose equivalent ranges and list of dose reduction techniques will aid in reducing collective dose equivalents from these dose-reduction targets. Knowing which components contribute to high maintenance or repair dose will aid in reducing routine maintenance collective dose equivalents. The radwaste dose reduction improvements will aid in reducing radwaste operations collective dose equivalent and reduce the number of radwaste workers who exceed their administrative dose limits. The identification and rating of manager's and workers' ALARA incentives will provide the basis for recommendations to improve dose reduction incentives. Lastly, the identification and rating of the key components of an ALARA program will aid in the development and coordination of the nuclear station ALARA programs.

The quality of the information gathered to date would not have been possible, were it not for the cooperation received during our nuclear plant visits and from our BNL-Industry Dose Reduction Advisory Committee. Recent presentation of our finding generated industry interest towards the BNL's ALARA Centers dose reduction efforts.[3-5]

In conclusion, if the good practices for dose reduction in our publications are put to use, this will result in enhanced nuclear safety, reduction of radiological risk and improved reputation of nuclear power in the United States. In addition, this would also represent another example of the NRC/INPO Radiological Protection Coordination.

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NEUTRON DOSIMETRY AT COMMERCIAL NUCLEAR POWER PLANTS

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This study is part of a larger program to evaluate personnel neutron dosimetry at commercial nuclear power plants. Previous measurements reported in NUREG/CR-1769 have shown that very few neutrons with energies above 1 MeV are found inside reactor containment. Although no "ideal" spectrometer exists, four types appear to be useful for reactor spectra measurements:

- multisphere or Bonner sphere
- activation foils
- proton recoil proportional counters
- ^{3}He proportional counters

All suffer from some deficiencies and do not cover the entire range of thermal to 1 MeV with great accuracy. Some are complex, difficult to use and often require complex unfolding codes. The ^{3}He spectrometer is a compromise between accuracy and ease of use. The ^{3}He spectrometer has three primary advantages:

- reasonable accuracy in the energy range of 50 keV to 1 MeV,
- an operating energy range extendable to thermal energies using neutron absorbing filters to determine approximate neutron spectra,
- demonstrated effectiveness in environmental chambers and the harsh environments of operating nuclear power plants.

The ^{3}He neutron energy spectrometer has three primary disadvantages:

- It is not an "off-the shelf" instrument--care must be taken in setting up a spectrometer system from commercially available components.
- The ^{3}He detector is very sensitive to low-energy neutrons--neutron absorbing filters and other techniques must be used to prevent pulse pile-up from giving erroneous data in the 20-keV to 700-keV range of energies.
- Only a crude estimate of the total flux density can be obtained in the energy range of 1 eV to 10 keV using a single cadmium filter.

Measurements were made at three pressurized water reactors and the drywell of one boiling water reactor during startup. Data from the ^{3}He neutron energy spectrometer indicate that more low-energy neutrons are present than are indicated by the multisphere spectrometer used at the same locations on the operating decks of the nuclear power plants. This

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difference has not been resolved, and it is recommended that additional measurements be made with activation foils or additional absorption filters over the ^{3}He detector to increase the accuracy of spectrum measurements in the 1-eV to 10-keV energy range. The dose equivalent rates measured with the ^{3}He spectrometer, a multisphere spectrometer, survey instruments such as the Snoopy cylindrical remmeter, and the tissue equivalent proportional counter for the reactor sites measured differed by a factor of two.

Results of previous tasks in the program indicate that portable remmeters used inside containment of nuclear power plants to determine neutron dose equivalent rates respond high compared to reference values. This high response demonstrates a significant dependence of the instrument response on the energy of incident neutrons.

A laboratory tissue equivalent proportional counter system (TEPC) was used to measure reference values for absorbed neutron doses and to determine reference values for neutron dose equivalents inside containment of commercial nuclear power plants. The TEPC was chosen as a reference measurement because TEPC measurements agreed closely with multisphere measurements which were the reference measurements for earlier subtasks, and because TEPC measurements agreed closely with calculated neutron dose equivalents produced by well-characterized neutron fields at the National Bureau of Standards and at the Pacific Northwest Laboratory Van de Graaff accelerator.

Routine neutron survey instruments were used to determine neutron dose equivalents inside commercial nuclear power plants and the results were compared to the reference values. The types of survey instruments included: (1) 9" remball, (2) 12" cylindrical remmeter and (3) a portable (2 lb) microprocessor controlled TEPC system. The prime disadvantages of the portable remmeters were: (a) weight, the polyethylene remmeters weighed 25 lbs each; (b) energy dependence, the polyethylene remmeters responded higher than the reference measurements by factors of 1.7 to 6 and (c) temperature dependence, which was a problem affecting the portable TEPC more radically than the polyethylene remmeters.

The study demonstrated that the TEPC is a superior technique to the portable remmeter for the accurate determination of neutron dose equivalent. Because the TEPC measures absorbed neutron dose directly and because the remmeters responded high, it was concluded that the TEPC is the superior technique for determining neutron dose equivalent/rate inside containment of commercial nuclear power plants and that the results from remmeters be adjusted to account for differences in neutron energy distributions, or that the remmeters at least be calibrated using D_2O-moderated Cf-252. The study also demonstrated that the characteristics of the laboratory TEPC do not allow for the simultaneous measurement of photon absorbed dose and that the temperature characteristics and readout of the portable TEPC available at this time limit its use to environments considerably less harsh than the environment encountered inside containment of an operating reactor.

ADEQUACY OF CURRENT SYSTEMS FOR MONITORING EXTREMITY EXPOSURES AT NUCLEAR POWER PLANTS

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Much of the present effort in extremity monitoring concerns itself with the response of dosimetry systems to calibrated beams under laboratory conditions. If one asks the question what extremity dosimeters should be measuring, a very different concern is highlighted.

The first step in examining the adequacy of current extremity dosimetry methods is to define what the dosimetry systems should be measuring. This step appears trivial until one considers that even the definitions of extremity and the corresponding dose limits vary. The current Code of Federal Regulations (10 CFR 20.201) does not explicitly define "extremity" except to set different dose limits to the "hands and forearms, feet and ankles." The Nuclear Regulatory Commission's (NRC) stance on the federal regulations is clarified in an information notice entitled "Dose Assignment for Workers in Non-Uniform Radiation Fields (US NRC Office of Inspection and Enforcement 1983). The NRC considered the "hand and forearm" to include the hand, the arm below the elbow, and the elbow. The arm above the elbow is considered part of the whole body. In this same information notice, the NRC has taken the position that the dose limit of "skin of the whole body" in 10 CFR 20.201 does not apply to the skin of the hand and forearm, but rather that the skin of the hand and forearm can receive doses up to the limit of 18-3/4 rem/quarter. An earlier information notice entitled "Clarification of Placement of Personnel Monitoring Devices for External Radiation" (US NRC 1982) implies that the knee, the leg below the knee, and the foot is classified as an extremity. Various agencies, such as the National Council on Radiation Protection and Measurements (NCRP), Nuclear Energy Agency (NEA), and International Commission on Radiological Protection (ICRP), have made recommendations concerning the definition and corresponding dose to the extremities. While the recommendations from the various agencies usually do not contradict one another, they do not form a clear consensus to guide the licensees.

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In order to answer the question of what dosimetry systems should measure, we realized that the definition and protection standards should be based on the ability of the component organs in the extremities to receive exposure to various radiations without pernicious effect. The working definition of extremities is the elbow, the arm below the elbow, the hand, the knee, the leg below the knee and the foot.

An analysis of the organs and tissues in the extremities indicated that the skin is most likely the critical organ for irradiation. Further, the basal or germinal cell layer of the skin is the most critical site. Whatever the results of the analysis, the licensee must comply with the letter and intent of the Code of Federal Regulations. The National Bureau of Standards Handbook 59 on which 10 CFR Part 20 standards were based, states that for the calculation or measurement of dose, "...the proper value is obviously the highest dose received by any skin area (on the order of a one square centimeter)". Therefore, the limiting dose is the highest dose averaged over one square centimeter at the basal layer.

Since 10 CFR 20 allows doses of 18-3/4 rem per quarter to the extremity and 3 rem per quarter to the whole body, the extremities can be limiting only in situations where the gradient in the radiation field exceeds 6 to 1 over distances of about one meter. Gradients of this magnitude can be obtained from beta sources and from spatially compact gamma sources. Typically, if a 6 to 1 gradient exists over one meter, much greater gradients exist close to the source.

Several dosimetry systems were evaluated at the Pacific Northwest Laboratory (PNL) calibrations facility. As expected, all of the systems responded well to broad beam gamma rays and to high-energy beta rays in low gradient fields. To investigate the effect of high gradient fields on extremity monitoring systems, irradiations were performed using one of the dosimeter systems and hand phantoms.

The dosimeter system particularly suited for the geometries we wished to study was a new "bandaid" type dosimeter. The bandaid dosimeter is composed of thermoluminescent material embedded in a carbon matrix under 4 mils of plastic. The active element has a density of approximately 5 mg/cm^2, which is similar to that of the basal cell layer of the skin. The relative thinness of the dosimeter simplifies the placement of the dosimeter and interferes minimally with dexterity. The response of the "bandaid" system was extensively studied in our calibration facility and its response to x-rays, beta rays, and gamma rays was as good as any system we investigated and often much better (see Table 1). This should not be construed as an endorsement of this particular dosimeter as a routine extremity monitor. This system has several features that would make its routine use undesirable: the dosimeters can only be used once, they are expensive to use, they fog the optics of the TLD reader, they are slow and time consuming to read out, and they are less sensitive than many other systems. For our research purposes, however, it was the system of choice.

Figure 1 shows the results of measurements made with the bandaid dosimeters on a hand phantom holding a sintered uranium oxide pellet. The dose rates at the thumb and index finger tips were over an order of magnitude higher than the dose rate measured on the palm side of the base of the ring finger (a common location for extremity dosimeters) and over two orders of magnitude higher than measured on the back side of the ring finger (where the dosimeter was shielded by the hand from the source).

These results are not exclusive for betas and other relatively nonpenetrating radiation, but also apply to high-energy betas and gamma rays. This effect arises almost exclusively from the spatial compactness of the source. Figure 2 illustrates the results of a hand phantom holding a cobalt-60 source. The dose rate at the thumb and index finger tips was over two orders of magnitude higher than it was at commonly used locations (such as inner and outer ring finger positions) for extremity dosimeters. The larger dose gradient was due in part to the distribution of cobalt-60 on the disk, which was not evenly spread arross the disk, but resembled a point source. Figure 3 shows dose rates for the hand phantom holding a lead cask containing the cobalt-60 source. The dose rates did not vary as much as for the bare cobalt-60 source. The highest dose was to the tip of the thumb and the ring finger tip. This was due to the position of the hand relative to the check source.

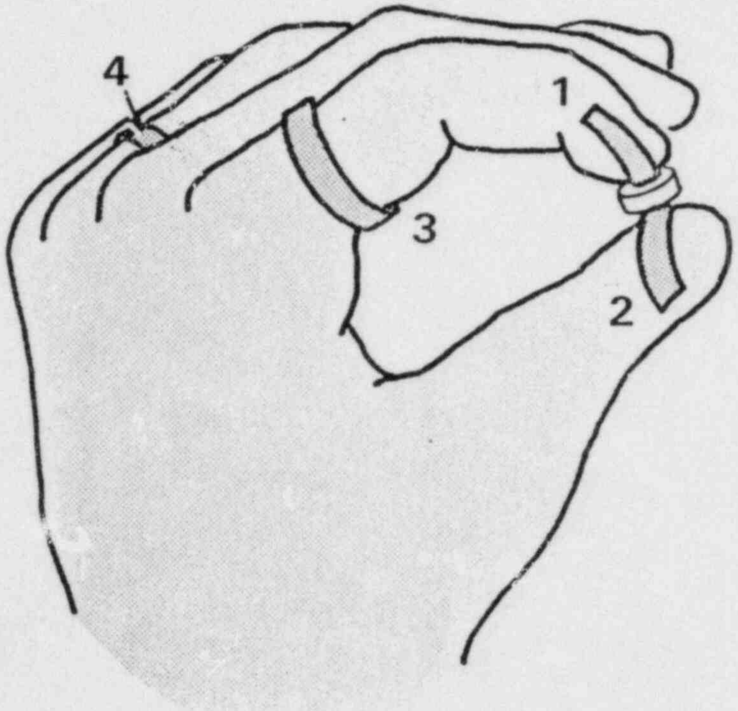
These laboratory irradiations indicate that when a compact source is handled, the dose delivered on contact is often several orders of magnitude higher than that commonly measured by extremity dosimeters.

In establishing whether gradients exist that would dictate the use of extremity monitors, measurements outweigh any theoretical consideration.

Table 1. Relative Response in Air (Uniform Field) Referencing Cs-137

Dosimeter Type	Beta		Photon
	^{85}Kr	$^{90}Sr/Y$	64 KeV X-ray
Chip	0.06	0.70	1.27
Bandaids	0.75	0.82	1.02

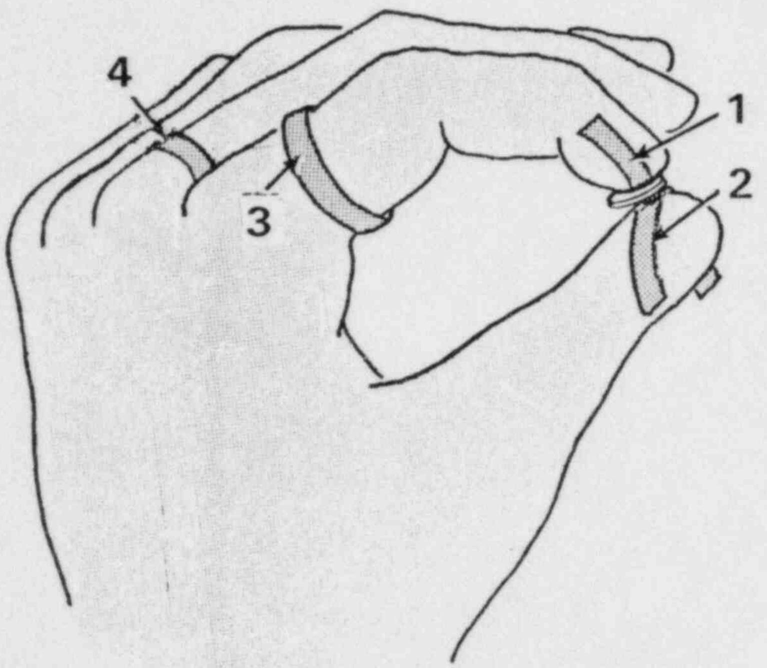
Extremity Exposure from a Uranium-Oxide Pellet



1. Index Finger-Tip (Contact)
   150 mrad/hr
2. Thumb Tip (Contact)
   150 mrad/hr
3. Index Finger-Front
   10 mrad/hr
4. Ring Finger-Back
   < 1 mrad/hr

FIGURE 1

Extremity Exposure from a Cobalt-60 Disk Source



1. Index Finger-Tip (Contact)
   5,000 mrad/hr

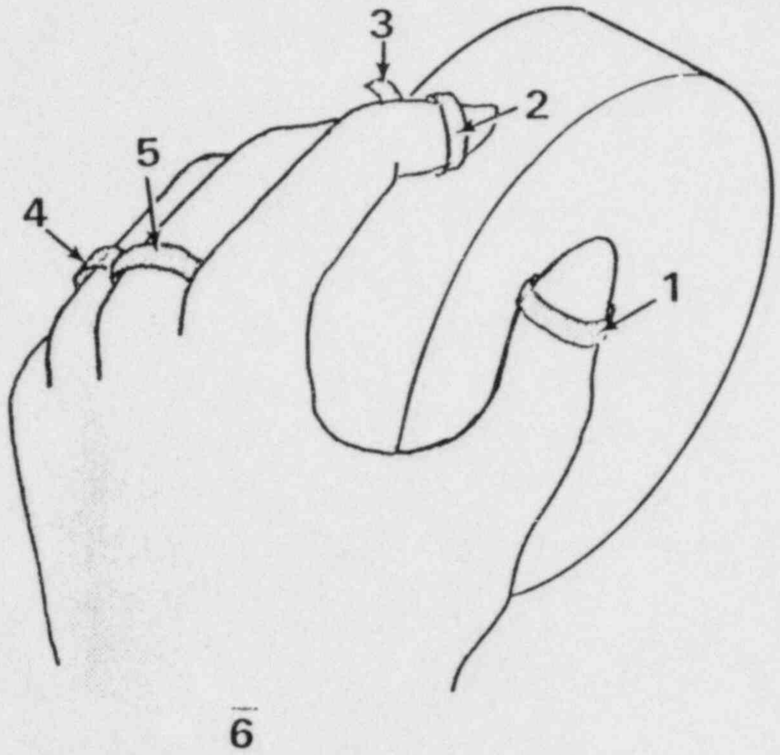
2. Thumb-Tip (Contact)
   5,000 mrad/hr

3. Index Finger-Back
   30 mrad/hr

4. Middle Finger-Back
   15 mrad/hr

FIGURE 2

Extremity Exposure from a Cobalt-60 Disk Source



1. Thumb-Tip (Contact)
   1800 mrad/hr

2. Index Finger-Tip (Contact)
   300 mrad/hr

3. Middle Finger-Tip
   300 mrad/hr

4. Ring Finger-Back
   200 mrad/hr

5. Middle Finger-Back
   70 mrad/hr

6. Back of Wrist
   70 mrad/hr

FIGURE 3

However, survey meters and dose rate meters (such as a portable ion chamber) underrespond as badly as dosimeter systems in high gradient fields.

The most commonly used instrument to establish dose rate levels is the portable ion chamber. Readings obtained during measurement of beams or spatially compact sources using ion chambers or other volume-averaging detectors are often interpreted incorrectly. The response of the ion chamber is determined by the number of ionization events taking place in the chamber averaged over the active volume. If most of the ionizing events are taking place in one portion of the chamber because of gradients in the radiation field, the indicated exposure rate will be less than the true exposure rate at the point of interest. Conditions that lead to gradients large enough to require correction of the ion chamber readings are not rare, and, in general, are the same conditions that make exposure to the extremity limiting.

When an ion chamber is used to measure contact exposure rates or exposure rates at short distances from a spatially compact source, there is a large ionization gradient across the chamber. Correction factors can be substantial for compact sources. The correction factors presented in Figures 4 through 6 assume a chamber having an active volume 5-11/16-inches long with a 3-inch diameter. Figure 4 shows the correction factors for various sized disk sources as a function of distance from the ion chamber. Figure 5 is a graph of correction factors for cylinders of varying length and diameter measured in contact with the front of the chamber (BNWL-MA-62).

The most radical corrections must be made for point sources. Figure 6 is a graph of the correction factors for point sources as a function of distance from the front face or perpendicular distance from the side of the chamber. The correction factor for point sources was calculated as the ratio of the dose at the given distance to a one-square-centimeter disk (shielded by 7 mg/square centimeter of tissue) over the volume-averaged dose measured by the chamber at the given distance from the face or side of the chamber.

Beta particles arising from surface contamination can easily create conditions that will produce critical gradients. For an isotope emitting beta particles with a maximum energy of one MeV, the bulk of the beta particles will be attenuated by the intervening air in less than three feet. The portable ion chamber is almost always used to determine the beta dose

Source Size and Distance Correction Factors for Disk Sources

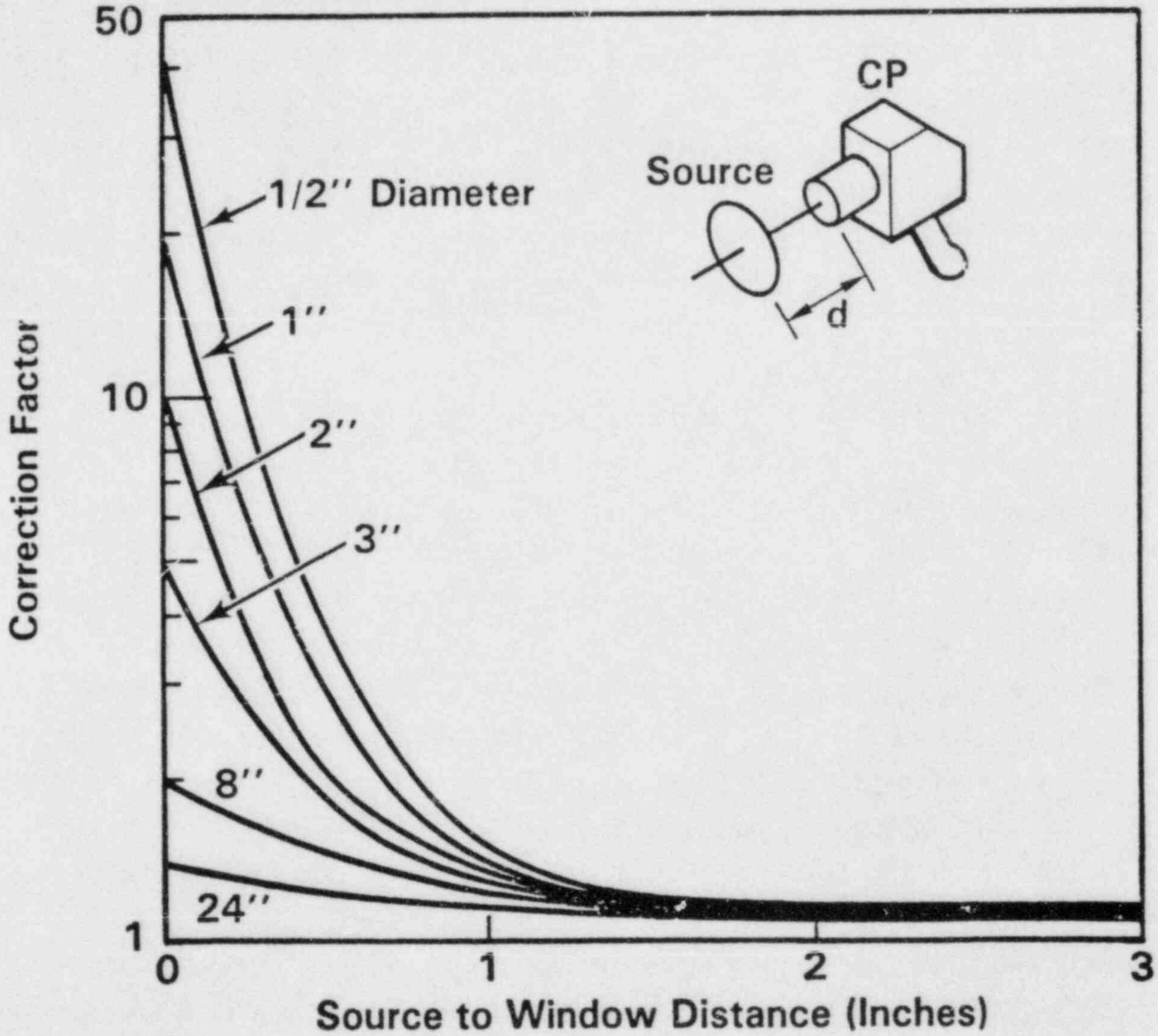


FIGURE 4

Correction Factors for Contact Dose Rates with Cylindrical Surface for Penetrating Radiation

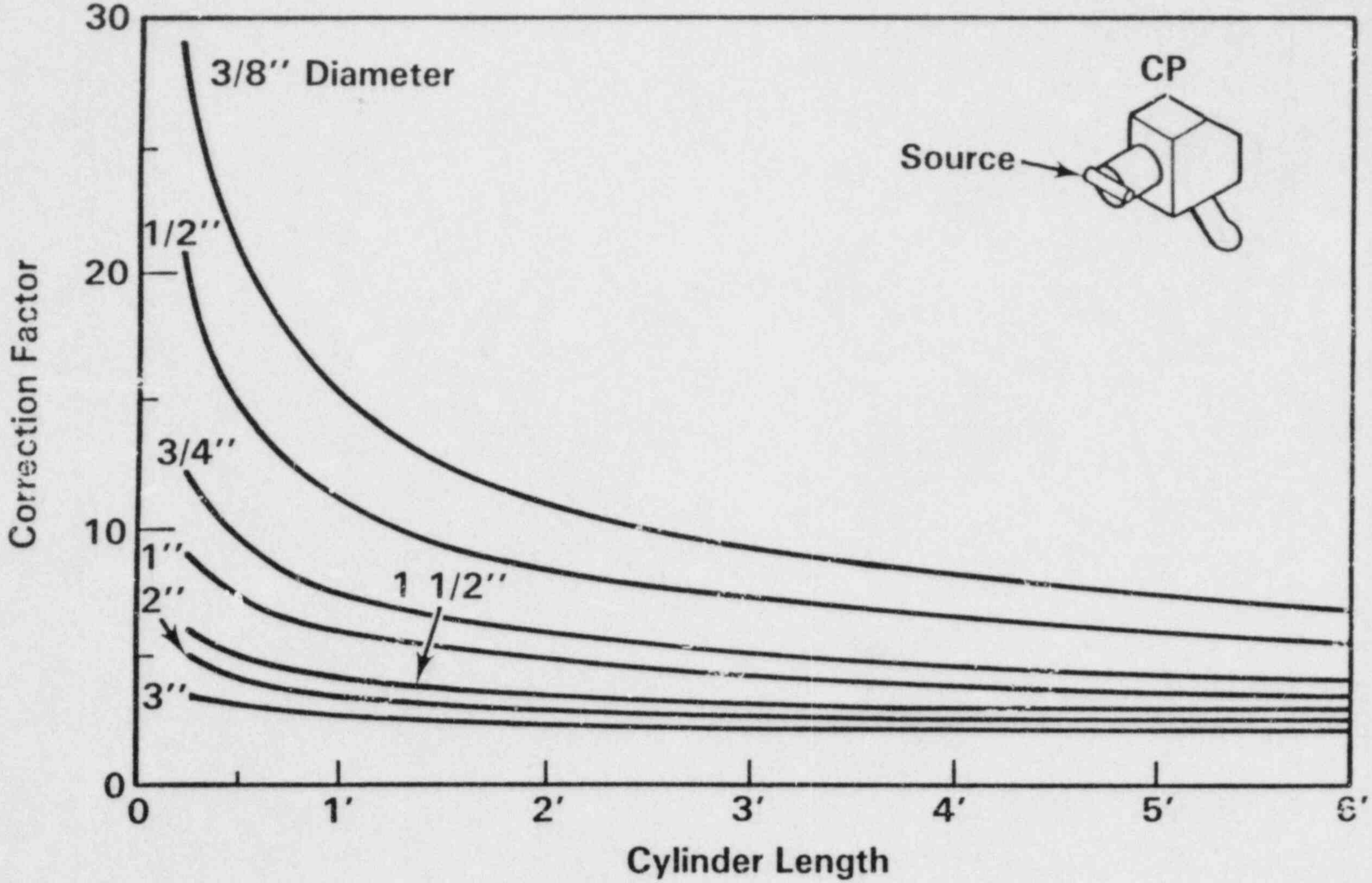


FIGURE 5

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Distance Correction Factors for Point Source

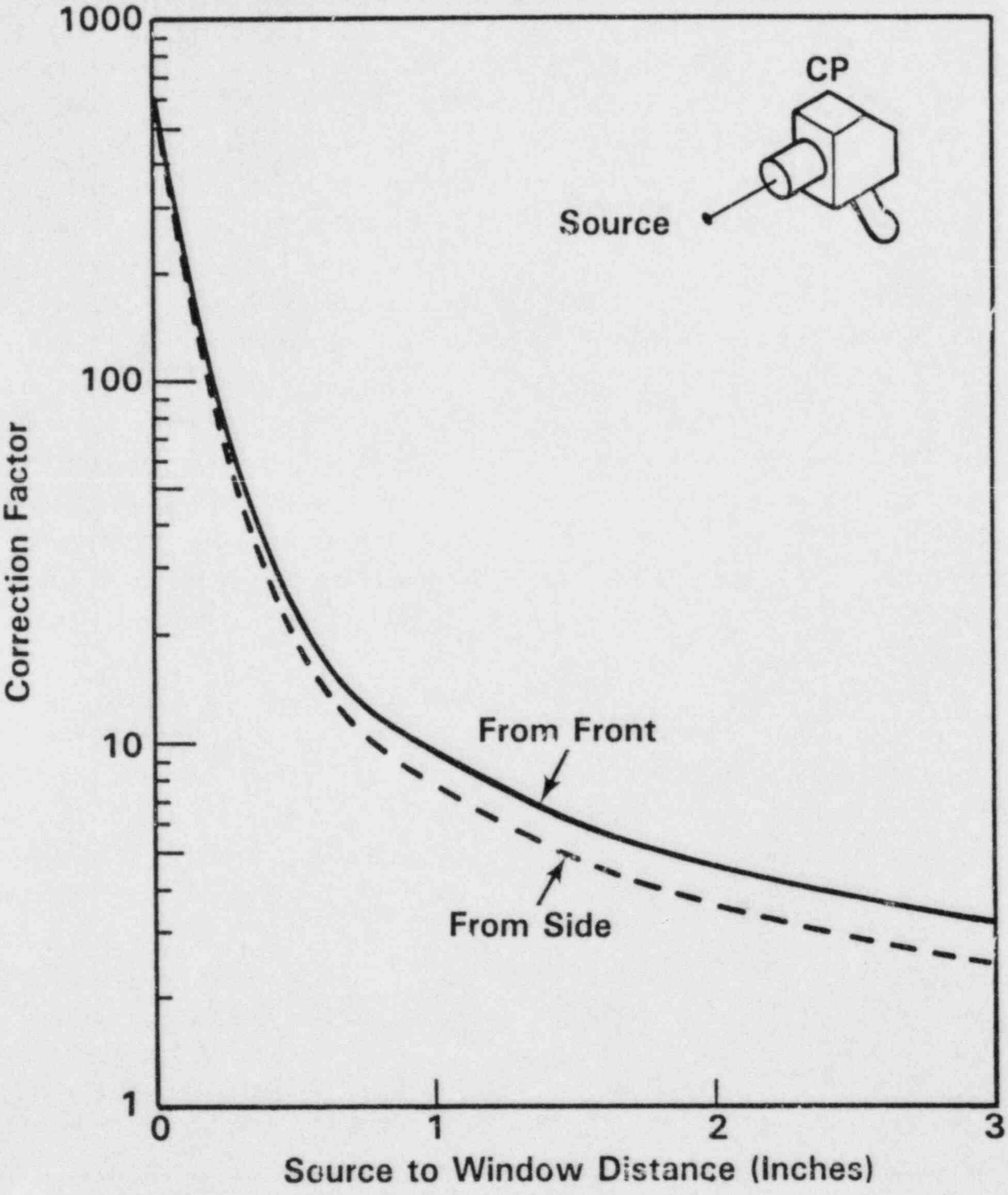

FIGURE 6

rate. All of the corrections discussed above for gamma rays (shown in Figures 4-6) emanating from spatially compact sources also apply to compact beta sources. Furthermore, since the dose rate is established by the difference between readings in "open window" (shield not in place on the front of the chamber) and "closed window" (shield in place) conditions, other corrections must be considered.

In the operation of commercial nuclear power plants, only a small portion of the radiation workers are limited by extremity exposures. Workers who are not extremity limited can be readily identified. Workers in fields generated by spatially large sources, where the measured gradients are less than six to one in one meter, should not need extremity monitors. Workers in jobs, such as channel head operations, where the ratio of extremity dose to whole-body exposure is known to be less than five to one should not need extremity dosimeters if adequate protection against beta radiation is provided. If personnel are controlled to less than three rem per quarter, then higher gradient fields can be allowed before extremity monitoring is required. For example, if in-house personnel are controlled to 2 rem per quarter, gradients of less than nine to one would not require extremity monitoring.

Workers who should be considered for extremity monitoring include almost any worker who handles a source smaller than about eight inches in diameter. Candidates for extremity monitors include radiation chemical technicians who handle samples of primary water, outage personnel involved in pump impeller or valve replacement, personnel handling sources during startup testing, and technicians using small sources for in-house calibrations.

In addition to the problems arising from the geometry of the source, many of the dosimeter systems used to monitor extremity exposures underrespond to low-energy beta particles (see Table 1). Components that have been in contact with primary water while the plant has been operating often exhibit beta dose rates of hundreds of rad per hour on their surface. If the gloves worn by workers replacing these components during outages do not provide completely adequate shielding for the high end of the beta energy spectrum, a small component of the beta radiation will penetrate the gloves and impart a dose to the workers' hands. This radiation will be of low energy and the dosimeter can underrespond by factors of ten or more.

Furthermore, if the dosimeter is worn facing away from the source, the dosimeter may show no response at all.

SUMMARY AND CONCLUSIONS

In general, only a small portion of workers at commercial nuclear power plants are limited by extremity exposures, and these workers can be readily identified. There seems to be no need for increased badging among the radiation workers. However, those workers who are extremity limited may not be receiving adequate dosimetry. For workers handling compact sources, unless contrary information is available, the tip of the thumb of the dominant hand can be assumed to be the limiting site, and dose to the thumb tip averaged over one square centimeter at the basal layer of the skin should be measured or estimated.

As discussed briefly in this paper, the assessment of dose in high gradient fields can be a difficult task. Radiation protection of the workers will be better served if the few workers who handle sources are closely monitored rather than wasting resources on general badging programs.

RECOMMENDED RADIOLOGICAL AIR SAMPLING AND INTERNAL CONTAMINATION CONTROL AT NUCLEAR POWER PLANTS

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Background

Federal regulations and guidance in controlling radiation worker exposure to airborne radioactive materials is based on the measurement and/or calculation of total intake of quantities inhaled or absorbed. Though bioassay and/or whole body counting are recommended as appropriate, these analyses are intended to be used for 'timely detection and assessment of individual **intakes** of radioactivity by exposed individuals'. In fact the common practice at Nuclear Power Plant facilities is to convert the quantity of (if significant) radioactivity measured in the body to the calculated amount which must have been taken in (inhaled) as the measure of compliance with the applicable regulations.

It has long been recognized by the NRC Technical Staffs that estimating the quantity of radioactivity inhaled by an individual worker involved large uncertainties. General air samples usually produce concentrations lower than those in the workers Breathing Zone (BZ). NRC guides have recognized this problem by specifying air monitoring programs which sample the Breathing Zone or concentrations known to be higher than that actually inhaled. In addition the availability of suitable samplers to obtain BZ samples and the practicality of requiring their use was somewhat in question to the NRC technical staff.

An NRC development contract was issued to provide a detailed review of the technical aspects of the problems and recommendations for practical upgrade of Federal guidance. This project accomplished a review of the nuclear industry experience and knowledge through a literature search, site visits to representative licensed facilities, telephone surveys of many others, laboratory testing of personal air samplers (lapel samplers) and aerosol diffusion experiments to verify key conclusions and assumptions.

Findings and Conclusions

The literature search revealed an extensive list of experiments and experience which indicated that general room or facility samplers measured concentrations which were up to two and three orders of magnitude lower than those measured close to the actual breathing zone of the individual worker. Diffusion experiments by EG&G verified the magnitude of the difference and in addition demonstrated differences of as much as a factor of three between BZ samples taken on the left and right shoulders of an individual worker. Though other experimenters have demonstrated greater consistency of multiple BZ samples on an individual worker when averaged over several days or weeks, the difference in individual samples can be expected to be as high as that measured in these experiments.

The inevitable conclusion is that air sampling to determine intake of the quantity of radioactivity by an individual worker is subject to large (several orders of magnitude) uncertainties. The ability to assure a 'conservative' sample is equally open to question.

Facilities which process large quantities of low specific activity materials (uranium mills) typically use a combination of general and BZ samples to calculate the MPC-hours exposure of individual workers, supplemented by urine analyses and annual whole body counts on the highest exposure potential personnel. Specific concentration limits in the urine provide a positive control but are generally not taken with a frequency sufficient to assure a definitive picture of individual and/or group experience. Whole body counts lack the sensitivity to be adequately definitive and little or no correlation with other exposure indicators was found.

Reactor facilities and other operations where mixed fission products and other beta-gamma emitters represent the major airborne radioactivity potential routinely use whole body counting as the preferred method of evaluating individual uptake. Though detection of a small fraction of the permissible body burden is practical, it is a common practice to convert a significant uptake measurement, using ICRP internal dose models, to a calculated amount inhaled as the measure of compliance. In reactor facilities air sampling is generally used to detect loss of control, spread beyond the control points, etc. and is not the preferred method of determining individual worker exposure. However, some facilities have reported using air sample concentration data multiplied by the time of exposure as the reported exposure even though whole body counting results were available also.

The conclusion based on experience and this technical review is that a direct measurement of uptake is preferable when practical. The number of assumptions (each with potentially large uncertainties) which must be made to determine uptake from air sample data make this method one of the least accurate and/or reliable.

It also appears unnecessary to convert a relatively accurate estimate of the amount of radioactivity in the body from direct measurements to an uncertain estimate of the amount inhaled by standard model assumptions simply for intake compliance purposes.

A detailed evaluation of commercial personal air samplers was performed to determine the capability and practicality of these devices to collect breathing zone samples. Many of the samplers tested performed reliably and well. This leads to the conclusion that personal air samplers are available and have sufficient capability to provide breathing zone samples of satisfactory reliability. However, personal air samplers are not widely or uniformly used. This is partly due to the concern that there would be a lack of sufficient sensitivity in some applications resulting from the relatively small volume of the PAS samples. Employee acceptance or rejection and the additional manpower needed for sample analyses, pump maintenance, etc. are also factors that result in a reluctance for widespread PAS use and must be considered in each program to evaluate their practicality.

Recommendations and Discussion

This investigation indicates the need to upgrade in three general areas:

1. Clarification of Federal regulations is needed to allow determination of internal dose in the most direct and accurate method for the situation as opposed to a measure of quantity inhaled.

2. Air sampling programs should be upgraded to better define their purpose as well as to improve the specific sampling techniques designed to meet these purposes.

3. Improve internal dose evaluation and documentation techniques.

Specific recommendations which follow logically from the results of this study are:

1. Current federal regulations should be changed to encourage internal dose evaluation by the most accurate technique. Evaluating the percent of intake and/or MPC-hr exposure is generally the least accurate for reactor facilities since there are many assumptions (each with potentially large uncertainties) which must be made to arrive at an internal dose and since whole body counting is a practical alternative. Current regulations do not require a dose calculation, just a calculation of the percentage of permissible intake. High activity in the air could result in negligible internal uptake based on particulate size, worker physiological differences, etc. The actual internal dose can and should be a practical limiting consideration.

2. Air sampling programs should be designed and used with the primary purpose of detecting loss of control, i.e. the presence of airborne radioactivity in the workplace. Plants and facilities handling radioactive material should be designed to contain the radioactive materials and prevent routine internal exposure to the workers or the public. High-volume constant (alarming) air monitors or frequently-changed passive samplers can provide timely information which will allow preventive, mitigative or corrective actions for control of internal exposures and prevent the spread of airborne contaminants to uncontrolled areas. The primary purpose of an air sampling program in a nuclear power plant should be to detect the presence of airborne contaminants outside the design enclosure or in high level work places thus providing for positive work place control.

3. The evaluation of internal dose should be made by the most direct method applicable. In nuclear power plant situations beta-gamma emitters represent the primary type of contaminant and whole body counting provides the most sensitive and least uncertain method of evaluating the quantities of contaminants in the body and for directly measuring the elimination patterns of the individual. It is pointless to convert the measured quantity present to a calculated quantity inhaled for purposes of compliance evaluation. The calculated dose to the organ(s) is an equally clear measure of compliance.

4. Personal Air Sampling (lapel) programs should be used to evaluate radioactivity concentrations in the work areas where substantial exposure potential is unavoidable. Results of general air samples are nearly always substantially lower than those in the air in the breathing zone. In some situations where pure beta or alpha emitters or low specific activity isotopes are handled whole body counting sensitivities may be insufficient to provide for adequate evaluation and control. In these cases breathing zone concentrations should be measured and bioassay programs strengthened (based on the results) to provide the sensitivity and maximum certainty of dose estimates. Personal Air Sampler use need not be continuous for each individual for operations where the air concentrations are not expected to vary widely. Periodic PAS vs general air concentration studies can establish a typical ratio between these two types of samples and allow routine use of less demanding area samples to provide necessary data for evaluation of intake values. Even when compliance with Federal regulations is judged on the basis of fractional MPC-hr exposures, the internal dose should be determined using bioassay techniques to verify the air monitoring indications.

Bioassay verification procedures should be used, even when whole body counting provides the estimate of the quantities present.

5. The evaluation and control of internal dose to workers from inhalation of radioactive materials in the workplace should be accomplished using a combination of the most appropriate techniques for each situation. The uncertainties involved in internal dose determination are large at best and several methods or sets of data should be used to evaluate the dose. Whole body (and/or organ) counting, bioassay and in some cases breathing zone and general air sampling data can be used together to assure the most reliable estimates of internal dose.

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Results of Comparative Assessment of U.S. and Foreign Nuclear Power Plant Dose Experience and Dose Reduction Programs

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Introduction

The objectives of this study were to determine how collective dose equivalents at U.S. nuclear power plants compare to those of other technically advanced countries, and to evaluate factors that contribute to the differences.

Fifty Health Physicists and nuclear engineers from 10 countries met at BNL May 29 - June 1, 1984 to exchange information and hold discussions on "Historical Dose Experience and Dose Reduction (ALARA) at Nuclear Power Plants".[1] Results of evaluation of data from this meeting and other data from recent publications are summarized here.

Gross Comparisons

Figure 1[2] shows gross comparisons of collective dose equivalent in severt/reactor/year for plants in Canada (Ontario Hydro), the United States, Japan, the United Kingdom (C.E.G.B.), France and Sweden. This comparison reveals that U.S. and Japanese plants yield average collective doses about 3 to 6 times higher than other countries under comparison. However, this gross comparison is insufficient for drawing conclusions since it does not properly reflect the influence of a number of important parameters such as: type of reactor, output in MWe, year of design or first commercial operation, and effective full power years of operation.

Type of reactor is important and this is somewhat indicated in this comparison since the Ontario Hydro plants are all pressurized, heavy water reactor (PHWR) systems and the U.K. plants are gas cooled reactor systems. The lowest dose equivalent per year was achieved at the gas cooled plants. The only U.S. gas cooled plant (St. Vrain in Colorado) has operated since 1974 at very low power levels (average to 1983 about 40 MWe compared to 330 MWe rating) and achieved similarly low collective doses when normalized by MWe generated.(3)

The Ontario Hydro plants use heavy water as coolant and differ considerably in plant design and operation. They provide a very interesting example of how dose reductions can be achieved if this is set as a high priority in both design and operation (this will be covered in more detail below).

The other countries represented on this Figure can be compared better by considering data in terms of type of plant (PWR or BWR) and by normalizing dose data by power generated (i.e. expressing results in terms of collective dose equivalent (rem or sievert) per unit electricity generated (MW-yr).

PWR Plants, rem/MW-yr

Figure 2 shows data for pressurized water reactor (PWR) plants. Rem per MW-yr is plotted vs. calendar year (1970 to 1983) for plants in the U.S.[3], Switzerland, Netherlands, the Federal Republic of Germany, Sweden, Finland and France. For most data three year averages were calculated and plotted to avoid large variations from year to year due to refueling cycles which are sometimes more than one year duration.

These data show U.S. plants averaging among the highest in terms of collective dose equivalent per MW-yr with the most recent Swedish plants (Ringhals 3 and 4) achieving the lowest values.

Reasons for differences are several: (1) the U.S. plants include older plants which are suffering correspondingly more steam generator tube failures. (2) The French have emphasized standardization of plant design which makes worker training more effective, since workers can go from plant to plant and work on nearly identical units. It also permits greater development and use of special tools such as steam generator manway cover handling devices, automatic eddy current testing machines and steam generator plugging machines. (3) At the Swedish units, great emphasis is placed on design, shielding, plant layout and careful control of primary circuit chemistry. By segregating and individually shielding highly active components, low dose rates during maintenance are possible. Also, low contamination levels in working areas minimize the need for respiratory equipment and attendant loss of worker efficiency. (4) Plant operators in Finland (Russian design) have achieved very low dose rates by carefully controlling primary water impurities, by avoiding high-cobalt stellite in primary systems, and by using larger steam generators, which have suffered relatively few tube failures, and which spread corrosion products over large surface areas.

Note that data for Swedish plants show progressive decreases in rem per MW-yr as newer plants are brought on line, thus indicating a strong and important "learning curve".

Further PWR comparisons can be made by referring to Figure 3 which shows rem/MW-yr vs. years of operation for U.S. plants of various sizes and ages compared to the Swedish plant (Ringhals 2) with the highest collective dose equivalent. Note that U.S. plants, which went commercial in '68 to '73, show the highest average doses (ranging from 0.38 to 2.3 rem/MW-yr); and three approximately 500 MWe plants (Kewaunee and Prairie Island 1 and 2), commercial in '73 and '74, show the lowest average doses (ranging from 0.08 to 0.5 rem/MW-yr) with larger U.S. plants and the Swedish plant being intermediate. The three plants with low doses had a total of only 29 steam generator tube defects (through 1980) or about 1/3 rd the average for post-'74 U.S. plants, whereas, the pre-'74 U.S. plants experienced an average of 904 defects per plant (through 1980). Thus, the number of steam generator tube defects is a major determinant of collective dose, as is well known. However, Beaver Valley, Calvert Cliffs 1 and 2, Davis-Besse 1, and Zion 1 and 2 plants also have experienced no tube defects through 1980 yet collective dose equivalents through 1980 were 1.0, 0.6, 0.24, and 0.6 rem per MW-yr, respectively, showing that other factors are also important and can cause large variations from plant to plant. It is also of interest that the three 500 MWe PWR plants with low doses all had Fluor Power Services, Inc. as architect engineers (but no other U.S. plants did).

BWR Plants, rem/MW-yr

Figure 4 shows rem per MW-yr vs. calendar year for BWR plants in Japan, the U.S.[3], Sweden, and Finland. Japanese plants show the largest doses. Data show an increase from 1.3 rem/MW-yr in 1972 to about 5.1 rem/MW-yr in 1977, followed by annual decreases to 2.1 rem/MW-yr in 1983. The Japanese experience reflects their emphasis on detailed and dose-intensive plant inspections and preventive maintenance activities during annual shutdowns. U.S. data for all plants compared to plants which went commercial in the '74-'79 period suggest some reduction in average collective dose equivalent per MW-yr, however, the improvement is small compared to the very impressive improvements shown by Swedish and Finnish BWR plants, which have shown progressive improvements from about 0.8 rem/MW-yr for the '75 Swedish plant (Ringhals 1), and about 0.25 rem/MW-yr for the '71 and '75 plants (Oskarshamm 1 and 2), to about 0.15 rem/MW-yr for the '75 and '77 plants (Barseback 1 and 2), and only 0.06 rem/MW-yr for the '81 plants (Forsmark 1 and 2). The two Finnish plants which went commercial in '79 and '81 (TVO I and II) fit the general Swedish pattern of progressively lower doses for newer plants.

Both the Swedish and Finnish plants have reactor systems designed by ASEA-Atom, a Swedish steam supplier which also acted as principal or contributing architect engineer on most of the plants. These plants have been designed with minimum cobalt content in primary systems surfaces, very careful control over primary water impurities and highly efficient reactor water purification systems. In general, to minimize introduction of corrosion products to the core, stainless steel with <0.05% cobalt or equivalent material is used for parts in contact with water which flows toward the reactor core. Therefore, most reactor internals, and water wetted surfaces in the primary system are made of stainless steel. Exceptions are minor parts such as springs, bolts, etc., which are made of nickel base alloys; and feedwater pipes outside the containment, feedwater heater housings, and end plates, which are made of carbon steel.(4)

U.S. BWR plants being designed are projected to have lower doses than currently operating plants. A factor of about two reduction is expected from design improvements (e.g. improved feedwater) and another factor of about 1.7 should result from source reductions due to more stringent materials selection criteria and more careful plant chemistry control.(5)

Data on dose vs. years of operation shown on Figure 5 illustrates that pre-'74 U.S. plants have experienced somewhat greater doses per Mw-yr than post-'74 plants. The trend for U.S. plants is generally upward for the first few years of operation, whereas, both the Swedish and Finnish plants have leveled off in about two years. The U.S. increases may reflect the larger contribution of cobalt-60 (with its several year build-up time constant) to doses in U.S. plants. The Swedish success may also be influenced by a goal of 0.2 rem/MW installed capacity suggested by the Swedish National Institute of Radiation Protection about 10 years ago. This is equivalent to about 0.3 rem/MW-yr generated, a very ambitious but apparently achievable goal.[6]

Dose vs. Plant Capacity for BWR & PWR Plants

Data on collective dose equivalent vs. rated capacity (MWe) [3] is shown on Figure 5 for U.S. PWR's and Figure 6 for U.S. BWR's. For PWR's the scatter in points (0.2 to 7.0) is large indicating any trend of dose with capacity is small compared to effects due to other factors. Scatter for BWR data points is less (0.9 to 7.4) and there appears to be a decrease with plant size for small plants (47 to 64 MWe) compared to those with capacity > 500 MWe.

Canadian Experience[2]

Ontario Hydro, the electrical utility for the Province of Ontario, employs pressurized heavy water reactors (HPWR's). Large collective doses received at their Douglas Point Nuclear Generating Station during 1967 to 1969 lead to a major effort at dose control during both design and operation. As a result a major commitment to dose control was made by senior management in 1970. Emphasis was placed on elimination of stellite (high cobalt content) alloys, addition of shielding, improvements in water purification systems, improvements in ventilation and air-drying systems (for airborne tritium control), and improved reliability and maintainability. The results are remarkable as shown on Figure 8. Collective doses per MW-yr were reduced from about 38 mSv/yr (3.8 rem/yr) in 1972 to about 3 mSv/yr (0.3 rem/yr) in 1981. During this same period U.S. experience at light water reactors fluctuated between 10 and 20 mSv/yr (1 and 2 rem/yr) per MW-yr generated with no apparent long term improvement.

An important aspect of the Canadian approach is the use of highly trained station workers for a major portion of all work. The number of workers per reactor has gone down from about 600 in 1970 to about 300 in 1982. During the same time period, the number of workers has increased from about 300 to about 1100 per plant at U.S. reactors. Station personnel now receive about 80% of the collective dose in Canadian plants compared to about 20% in U.S. plants. This difference in workforce complement is believed to be an important element in the Canadian success.

Conclusions

Based on data evaluated to date it is clear that U.S. plants have higher collective dose equivalents per reactor and per MW-yr generated than most other countries. Factors which contribute to low doses include: 1) minimization of cobalt in primary system components exposed to water, 2) careful control of primary system oxygen and pH, 3) good primary system water purity to minimize corrosion product formation, 4) careful plant design, layout and component segregation and shielding, 5) management interest and commitment, 6) minimum number of workers and in-depth worker training, 7) use of special tools, and 8) plant standardization.

It should be pointed out that reductions in exposure are more difficult and costly in plants already built and operating. The cost-effectiveness of dose reduction efforts at U.S. plants should be carefully evaluated before recommendations are made concerning existing plants. This is the subject of a related on-going study at BNL which should be completed in the near future. Important research projects on cobalt source identification, optimum primary system chemistry, primary system decontamination and surface pre-treatment (passification) to prevent corrosion and deposition are currently being sponsored by the Electric Power Research Institute. Results from these studies will be extremely important in aiding future efforts at dose reduction and control.

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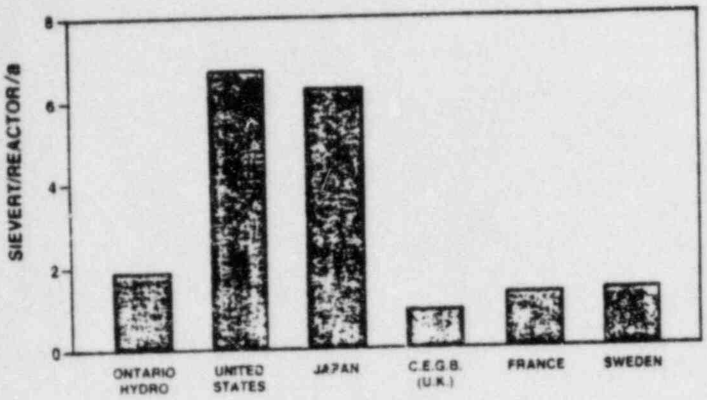
Fig. 1.
COLLECTIVE DOSE PER REACTOR PER YEAR
5 YEAR AVERAGE 1978-1982(2)



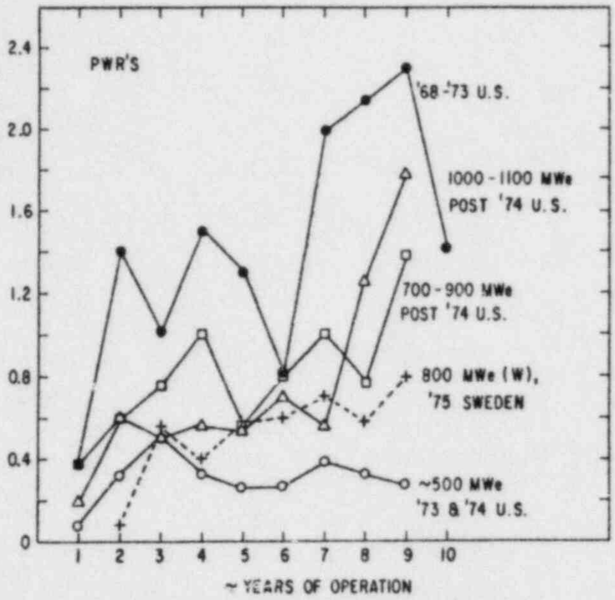


Fig. 3. Collective dose equivalent per MW-yr vs. approximate years of commercial operation for PWR plants.

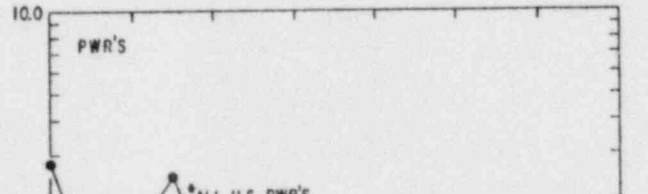


Fig. 2. Collective dose equivalent per MW-yr vs. calendar year for PWR plants.

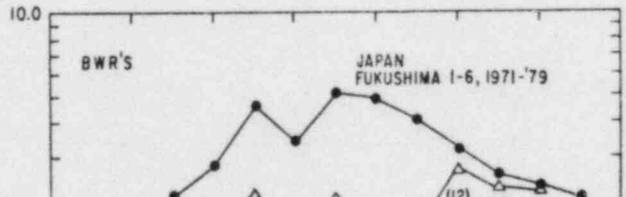


Fig. 4. Collective dose equivalent per MW-yr vs. calendar year for BWR plants.

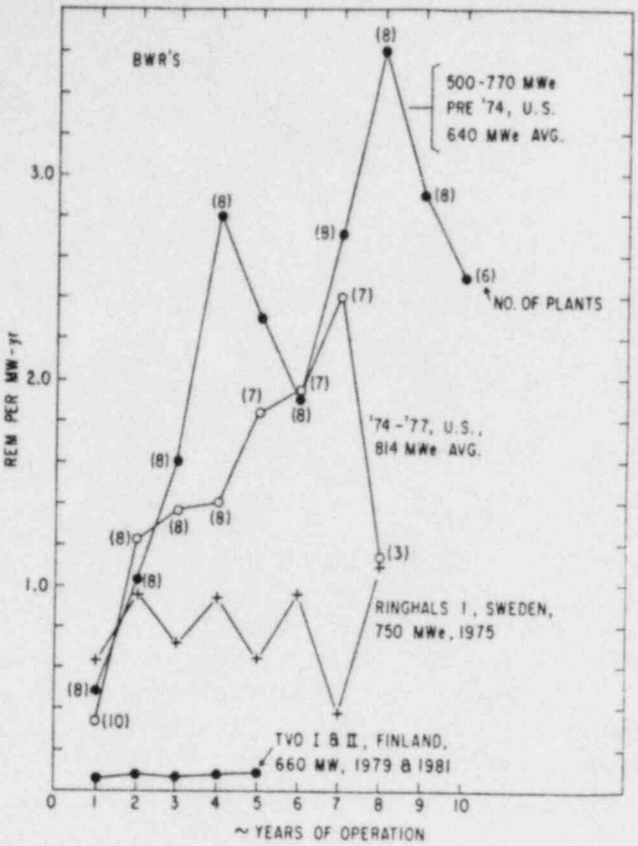


Fig. 5. Collective dose equivalent per MW-yr vs. approximate years of commercial operation for selected BWR plants.

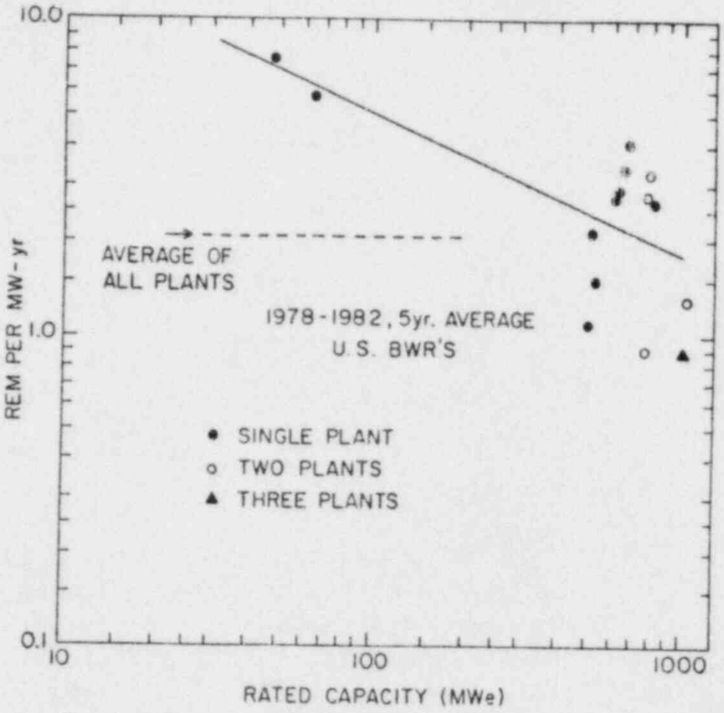


Fig. 7. Collective dose equivalent per MW-yr vs. rated capacity for U.S. BWR plants.

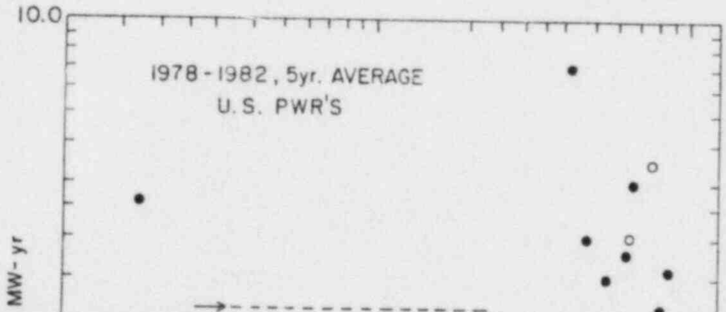


Fig. 6. Collective dose equivalent per MW-yr vs. rated capacity for U.S. PWR plants.

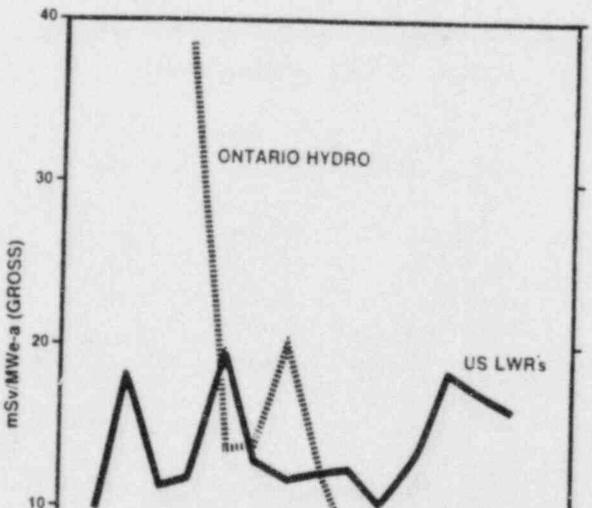


Fig. 8. Collective dose equivalent per MW-yr vs. calendar year for Ontario Hydro PHWR plants and U.S. LWR plants.[2]

DEMONSTRATION PROJECT FOR ROBOTIC INSPECTION SYSTEMS AT NUCLEAR POWER PLANTS

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ABSTRACT

A cost-effective approach for applying industrial robotics technology to nuclear power plant inspection work was developed during a Phase I study. High potential areas for inspection robots were identified at a BWR and PWR plant. Phase II of this NRC project includes the design, fabrication, and demonstration testing of a surveillance robot (SURBOT) at the Browns Ferry Nuclear Plant. SURBOT will replace workers in performing inspections and radiation mapping within radiation controlled areas.

INTRODUCTION

The nuclear industry is actively seeking ways to reduce the radiation exposure of workers at commercial power plants to as low as reasonably achievable (ALARA) levels, as recommended by the NRC Regulatory Guide 8.8, and to maintain the economic viability of nuclear power. In reviewing the literature, it is apparent that no single technique can be cost effectively applied at existing plants to meet the ALARA guidelines.[1] Rather, a variety of occupational radiation exposure (ORE) reduction techniques have been and are being applied. These are grouped into two basic approaches: (1) reducing the radiation level in work areas, and (2) reducing the time the workers must spend in radiation areas.

It is generally agreed that a primary cause of exposure is the radioactive corrosion products that accumulate on equipment and building surfaces external to the core. The accumulations create radiation fields in areas where workers must perform surveillance and maintenance work. Major programs are underway to reduce these radiation levels by preventing the formation of corrosion products, filtering or otherwise preventing the deposition of corrosion products in plant components, and remotely decontaminating plant components to remove deposited material.

Reducing the time workers spend in radiation fields is being accomplished in a number of ways such as the improvement of component reliability and maintainability. Good examples of the need to improve component reliability is the fact that valve failures account for about 19% of all LWR shutdowns,[2] and that steam generator inspection and maintenance accounts for a nominal ORE of about 135 man-rems annually at PWR's.[3] Improved component maintainability includes the use of quick disconnects for piping and electrical cables, easily replaceable pipe insulation, power assembly/disassembly tools and automatic machines for weld testing, pipe cutting, and valve seat refinishing. Most plants have on-going worker training programs and dedicated ALARA coordinators to decrease plant outage periods and reduce exposure levels.

Another means for reducing exposure is to use remote techniques for inspection and maintenance work tasks. Phase I of this project was a feasibility study to determine whether robotics can replace workers in radiation areas for certain jobs and be cost/beneficial to the power plants. The scope of the study was limited to surveillance and inspection activities and only considered the retrofit of robotics into existing power plants. The Electric Power Research Institute is evaluating robotic applications for maintenance tasks.[4] The results of Phase I were positive,[5] and NRC has authorized Phase II which includes the design, construction, and demonstration testing of a surveillance robot at the Browns Ferry Nuclear Plant.

CURRENT POWER PLANT ROBOTICS

Replacing workers with remotely operated or robotics-type equipment is not a new idea. Generally, this effort has been directed towards high radiation tasks at plants in the U.S. and other countries. The most comprehensive program for applying robotics to nuclear power plants appears to be in Japan. Their stated objectives are to reduce exposure as well as manpower and plant downtime. The accomplishments to-date include the construction and plant installation of remotely operated and programmable equipment for refueling, cask decontamination, inspection within primary containment, an inspection machine that roves through building areas following a floor tape, pipe welding, control rod drive replacement, and ultrasonic flow detection of primary components.[6] Most of this equipment has been developed for BWR's by the Toshiba and Hitachi Ltd. Corporations. Meetings were held with engineers from these companies and the Tokyo Electric Power Company (TEPCO) in December 1983. They generally believed that the robotic systems function as intended and do reduce exposure. However, no specific data were available for a cost/benefit analysis. They described a number of other power plant robotic applications that are currently under consideration or development and funded by the Japanese government and utility companies. Examples include an automated laundry for contaminated protective clothing; portable probes for the internal visual inspection of pipes, tanks, and valves; walking transporters that can carry inspection equipment and robot arms; and robot arms that can be operated either by computer or teleoperated with a master arm.[7]

The U.S. plants, with only minimal Government or utility-sponsored programs, are very cost-conscious in the application of specialized inspection/maintenance equipment. As a result, the applications have been limited to using semi-remote equipment as a means of reducing exposure. These are single function machines that are installed and removed manually by workers but operate automatically after installation. A study to evaluate the use of robotic systems for nuclear plant maintenance was recently performed by the Electric Power Research Institute (EPRI). It identified and analyzed a number of high exposure tasks with recommendations that further development be implemented for robotics to clean the reactor cavity after refueling, perform health physics surveys in radiation areas, and unbolt large pipe flanges. Positive cost benefit evaluations were presented for these applications to justify the development.[4]

A further conclusion in the EPRI study was that commercially available, industrial robot systems were not directly applicable to reactor maintenance without extensive development work. This same conclusion was reached in a reactor maintenance study performed by Catalytic, Inc. for the U.S. Department of Energy. It categorized remote surveillance/diagnostics and robots as advanced technology with a projection that remote surveillance would be widely adopted by the year 2000, but full development of robotics would not occur before 2020.[1] The Westinghouse Corporation surveyed commercial robots, concluded they were inadequate, and developed a remotely-operated service arm (ROSA) for performing work at power plants. The ROSA was recently used to inspect and repair a steam generator at the Zion nuclear plant, however, Westinghouse classifies it as a general purpose tool positioner that can perform a variety of service tasks.

Roving robots are seriously being considered for performing remote surveillance and limited maintenance tasks. EPRI is currently sponsoring the development and testing of a vehicle that can be remotely maneuvered through complex paths including stair climbing. It is to be equipped with television viewing; sensors to measure temperature, vibration, radiation, and sound; and a robot arm for pick/place tasks.[8] A functionally similar system has been developed and used in a BWR plant in Japan; however, it follows a fixed path (floor tape) and cannot climb stairs.[9] Walking machines are also being considered by EPRI in that they could conceivably have more versatile mobility than floor rolling systems.[10]

According to revised estimates, cleanup workers at the Three Mile Island, Unit 2 reactor are likely to receive a total collective radiation dose of between 13,000 and 46,000 person rems for the entire cleanup project. This report states that robotic technology could reduce the worker dose by about 40% but that the state of robotic technology is a long way from producing the type of versatile devices that are needed.[11]

CARA METHODOLOGY

After touring a number of power plants and having discussions with plant personnel, it became apparent that a variety, or family, of robotics systems would be needed to perform surveillance and inspection work. Selecting sensors that can perform the required tasks remotely is straightforward. However, installing them into the existing plants is complicated by a number of conditions including the following:

- Each plant contains a large number of radiation areas (rooms) that require an approved permit for entry.
- The radiation areas are located at different elevations within the reactor and auxiliary buildings.
- Reasonable access is provided for equipment transport to most radiation areas, but some are even difficult for a worker to enter.
- There is no defined aisle around much of the equipment and piping within most radiation areas, especially in the lower containment areas of PWR's.
- Suit-up rooms, monitor stations, and/or step-off pads are located at the entry to most radiation areas to minimize tracking of contamination into the hallways.

- Tasks within some of the areas require workers to climb stairs, vertical ladders, spiral staircases, and to maneuver under/over pipes and other obstructions.

- Some concern exists regarding the use of wireless signal transmission using radio frequencies within the containment building.

- Some areas require entry for surveillance every shift while others are entered monthly or less frequently.

- Many of the areas, especially in PWR's, have floor obstructions such as water dams, pipes, and ducts.

- The surveillance of some areas must be completed quickly, when on critical path for plant outage, and a team of workers is deployed to do the work.

- Routine surveillance during power operation is performed simultaneously with a number of worker teams because of the many areas involved.

- Many of the areas have very poor lighting, few electrical outlets for power supply connections, and restrictions on the addition of wall penetrations.

It was decided that the best way to cope with this complicated situation was to approach plants on an individual basis and determine the feasibility of robotics by performing a systematic area-by-area analysis. This methodology is called the cost-effective approach for robotics application (CARA) as shown on Figure 1. The left column describes the activities to be performed at a specific plant, and the sequence of activities for designing a robotic system is shown in the right column.

There are a number of factors that can determine the cost effectiveness of a robotics system and will only be revealed by an in-depth review of a plant's operating experience. ORE data records have limited value since most systems in use are directed at recording and tracking of exposures by individuals. A few of the more advanced computerized systems provide dose breakdown by system, component, jobs, and plant areas. More definitive data can be obtained by reviewing the entry permits and the radiation survey maps for specific plant areas.

Man-hour reduction is a possibility with robotics, and detailed breakdowns should be obtained for entries into specific plant areas. It should include all personnel involved in the entries and the time required for suit-up. A point to note is that a health physics survey of areas is now required for worker entry to perform surveillance/inspection but is not required if these tasks can be performed remotely. The records for time periods when the plant was shutdown or operated at reduced power should be carefully reviewed to determine the specific plant areas and activities that were involved. It may be possible to install robotic systems in these areas to minimize the need for power level disruption. Other factors that should be reviewed are worker safety and the amount of protective clothes, scaffolding, tool disposal, waste disposal, etc., that result from personnel entry into specific areas.

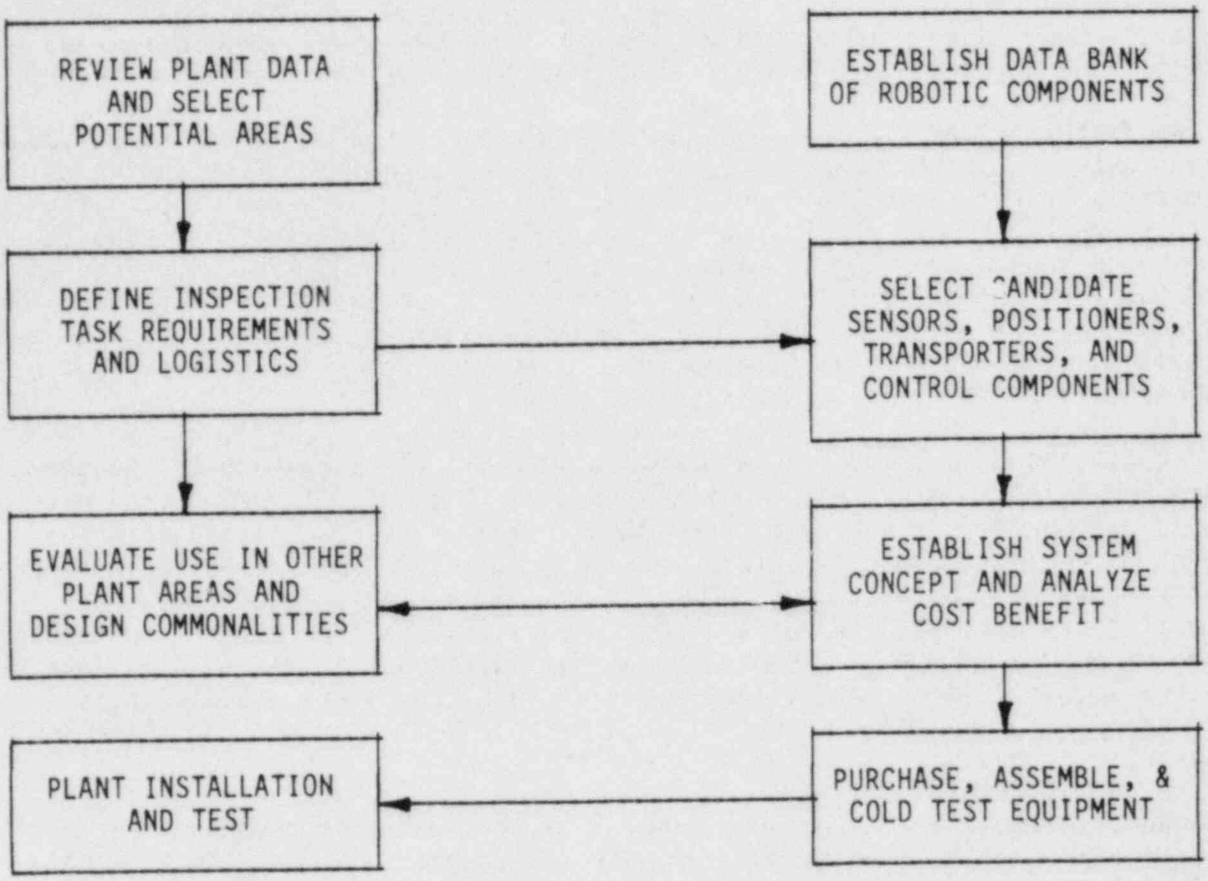

Figure 1 Cost-Effective Approach for Robotics Applications (CARA).

Information obtained from evaluation of worker exposure should be compiled and categorized by specific plant areas after which the areas can be prioritized according to potential benefit using robotics. Selecting robotic components and completing the design of a system requires specific plant information. The surveillance specifications for each area must be categorized including type, accuracy, frequency, etc., in order to select sensors.

Working conditions and space envelopes for each area must be defined to select positioners and transporters for the sensors. That is, the physical location of each inspection point must be determined, obstructions and access space to the inspection points defined, and access limitations to the area identified. Environmental conditions within the area are needed to establish the radiation, temperature, and moisture resistance requirements of the equipment components.

Special attention should be given to establishing the requirements for control and data acquisition systems as they can have a significant impact on the cost of a robotic system. In cases where an area inspection is performed on a non-routine basis, it may be acceptable to have switch-operated controls and data display located outside the access door into the area. The worker could manually operate the robot and record data without entering the area. For routine inspections it may be desirable to apply more sophisticated systems, perhaps even to the extent of operating the robotic system from the reactor control room. The most cost-effective guideline would be to use the least complicated control system modules that meet the specific inspection needs.

A key factor in the CARA methodology is the data bank of information on robotic components. This should be a comprehensive product information listing of commercially-available equipment components that have potential application to inspection tasks. It should include sensors, positioning arms, transporters, and control/data components. The objective is to treat these components as modules that can be selected and arranged into a configuration to perform the surveillance/inspection tasks within specific plant areas. The data bank should be updated by entering new products as they become available and removing those which perform unsatisfactorily.

There will be a number of areas within a specific plant that could use the same, or slightly modified, robotic systems. It should, therefore, be possible to move a system between areas for common usage. The removal and transfer might be accomplished by remotely controlled means in some cases but more probably by workers because of access limitations and contamination control requirements for the plant areas. For plants that have more than one reactor, the common use of specialized equipment for infrequent inspections can greatly increase cost effectiveness. After establishing a system concept, it is necessary to prepare a cost-benefit analysis and, if approved, purchase and assemble the equipment and perform in-plant testing.

In summary, the CARA methodology recognizes that existing nuclear power plants are comprised of a large number of individual rooms or areas. Each is unique relative to the type of surveillance/inspection performed, the costs and hazards involved with personnel entry, the arrangement of equipment and piping, and the access space available for robotics to operate. Solutions for each area are based upon the modular arrangement of commercially-available sensors and other robotic components to minimize purchase and installation cost as well as maximize reliability. It is expected that a family of robotic systems will result from this effort which will be applicable to other nuclear plants.

SURVEY OF POWER PLANTS

Three Tennessee Valley Authority (TVA) nuclear power plants were surveyed to identify specific plant areas where there is a high potential for surveillance/inspection robotics and to demonstrate the CARA methodology. The Sequoyah and Watts Bar plants are both two-unit PWR's and are very similar in design. They were included in the survey because Sequoyah is in operation whereas Watts Bar is in the final stages of construction. The objective was to identify specific problem areas and ORE levels at Sequoyah and then to thoroughly evaluate the high potential areas or rooms at Watts Bar where free access was permitted without radiological concerns. The reactor units at Sequoyah are 1148 MWe each, and Unit 1 began operating in July 1981 and Unit 2 in June 1982. The units at Watts Bar are 1175 MWe with Unit 1 scheduled for startup in September 1984 and Unit 2 in 1986. The reactors at both plants were supplied by Westinghouse with TVA as the architect-engineer.

The first step in the survey was to obtain detailed plant drawings from TVA and become familiar with the arrangement and function of equipment and rooms within the plants. This was followed by visits to each plant and guided tours through the areas which are, or will become, radiation areas. Protective clothing was required for some areas of the Sequoyah plant tours. Detailed discussions were held with plant personnel including the ALARA coordinator, health physicists, operators, and maintenance workers.

The ALARA coordinator described the computerized ORE management system that was implemented at Sequoyah in 1983. It is versatile in that ORE data can be sorted and printed out according to plant area, work function, craft, individual, entry data, plant system, etc. The system involves the use of a radiation work permit (RWP), an RWP timesheet, and an area survey map. The RWP is an authorizing document that establishes the requirements for entry and work within a specific radiation area. Before issuing an RWP, the HP technician must obtain a good working knowledge of the job location and equipment, the type of work to be performed, and the current radiological hazards in the area.

Determining the radiological hazards requires that HP technicians enter the area and perform a radiation and contamination survey. The results of the survey are placed on an area survey map and serve as a basis for defining the RWP entry requirements. The survey of areas that are entered infrequently is usually performed prior to entry. Areas that are entered frequently are surveyed on a weekly or daily basis. Both the RWP and survey map are posted outside the access door to the area. The RWP timesheet provides most of the detailed data that is entered into the computerized ORE program. It is also posted outside the access door to an RWP area. Each worker that enters the area records his name, craft, social security number, time in, time out, and dosimeter reading. It was estimated that about 10,000 RWP timesheets were issued in 1983. Categorization of the RWP timesheets is achieved by assigning unique identification numbers to areas and equipment in the plant.

The Sequoyah plant workers involved in this study were asked to recommend candidate areas and tasks for using surveillance/inspection robotics. Their response was mixed in that exposure reduction was a driving force for some but others were concerned with worker safety, high (and low) ambient temperatures in some areas, and the need to suit-up for an inspection entry that requires only a few minutes time. The recommended areas are listed in Table 1 along with specific data obtained from the ORE management system. It is interesting to note that the Sequoyah plant has only been operating for a short time, and the input from more mature plants (Zion, Oconee, and McGuire) shows background radiation levels in the candidate areas is about the same as Sequoyah. The selection of high potential areas was based on the ORE and man-hours required for inspection, worker safety concerns, and a detailed examination of the specific areas at the Sequoyah and Watts Bar plants. Plant personnel described the specific inspection to be performed in each area, location of inspection points, contamination control requirements, access into the area, and obstructions within the area.

The Browns Ferry Nuclear Power Plant, operated by TVA, is a three-unit BWR located in Decatur, Alabama. The architect-engineer was TVA and the reactor-supplier was General Electric. Unit 1 began operation in August 1974, Unit 2 in March 1975, and Unit 3 in March 1977. Each unit is capable of 1067 net MWe and is equipped with independent cooling systems, turbines, generators, and control rooms. The three units share a common radwaste building, service building, and offices. Detailed plant drawings were obtained from TVA to become familiar with the arrangement and function of equipment and rooms

Table 1 Candidate Sequoyah Plant Areas

No.	Area or Room	*Total Entries Man-Rems	*Total Entries Man-Hours	*Inspection Entries Man-Rems	*Inspection Entries Man-Hours	*Back-ground mR/hr	Frequency of Inspection Entry
1.	Upper Ice Condenser	19.59	15,753	4.61	2,590	2-5	Daily
2.	Lower Ice Plenum	1.79	1,034	0.53	32	2-800**	Infrequent
3.	Keyway	1.58	180	0.03	4	700-200,000	Refueling
4.	Upper Air Lock	0.26	1,113	0.08	328	1-2	Daily
5.	Lower Air Lock	0.75	2,972	0.07	301	1-2	Daily
6.	Waste Package Room	3.50	1,643	0.67	212	20-800	Daily
7.	Spent Resin Loading Area	5.50	367	1.04	127	1000-20,000	Monthly
8.	Reactor Coolant Pump Motors	19.93	2,482	2.56	251	1000-5000**	Monthly
9.	Steam Generators	84.98	4,899	31.54	1,458	100-40,000	Refueling
10.	Under Reactor Vessel Head	25.92	1,888	8.80	223	5000-15,000	Refueling
11.	Seal Table	2.86	1,697	2.03	1,097	2-5	Daily
12.	El. 690 Pipe Chase	5.93	1,095	0.42	102	2-400	Infrequent
13.	El. 669 Pipe Chase	1.64	172	0.23	19	5-300	Infrequent
14.	El. 653 Pipe Chase	1.69	203	0.08	7	2-250	Infrequent
15.	Valve Galleries	14.31	4,359	1.42	550	2-1000	Infrequent

*Values are based on January-November 1983 data for Units 1 and 2 combined.
**At Power

Source: Sequoyah ALARA Coordinator

within the plant. This was followed by a visit to the plant and a guided tour through the radiation areas. Discussions were held with plant personnel including the ALARA coordinator, health physicists, operators, and maintenance workers.

Browns Ferry's exposure management system is controlled by the use of special work permits, special inspection permits, supplemental data sheets, and area survey maps. A computerized program maintains a record of the exposure received by every individual entering a special work permit or special inspection permit area. The ALARA coordinator, shift engineer, and the Health Physics Department supervise the exposure management system. The computerized ORE system at Browns Ferry provides data printouts by system, component, and craft. It does not track exposure or man-hours by area as the Sequoyah plant system did.

Meetings were held with the health physics personnel and the ALARA coordinator to obtain their recommendations for candidate areas that would be most beneficial for the application of robotics. A total of 21 areas were identified at the meeting as shown in Table 2. Estimates were obtained for each area including background radiation level, frequency of entry for inspections, number of workers that enter, and the ambient room temperature. Most of the inspections were for visual surveillance of pipe and valve leaks and to measure radiation levels.

During the plant tour, it was apparent that there is vastly more access space for robotics in a BWR than a PWR plant. It was also noticed that the radiation level of pipes and components is much higher at a BWR, which makes the justification of robotics for ORE reduction more possible. Further, there are numerous possibilities for common usage of robotics systems between rooms that have a similar configuration and inspection sensor coverage requirements. As discussed previously, prior to worker or surveillance/inspection personnel entry into these areas or any SWP area, an HP technician must perform a radiological survey. It consists of gamma monitoring the general background and the specific components that are to be inspected or maintained. Contamination smearing is also performed in the area. The radiological data is recorded on the survey maps which are posted outside the entrance to each area.

ROBOTIC APPLICATIONS

The survey of power plants clearly indicated that retrofitting robotic surveillance/inspection systems into existing plants will be difficult. Selecting commercially-available sensors that can replace a worker in performing visual surveillance, measuring radiation levels, etc., is not a major problem. But, duplicating the agility of a worker in transporting and positioning a sensor within the large variety of plant radiation areas is a formidable challenge.

Robotic solutions were proposed for some areas and include an estimate of the cost to implement and the cost of benefits derived. The benefits were based on TVA costs which included: \$20/hr for health physics technicians and operators, \$12/hr for unskilled labor, \$1000 per man-rem, \$5 for a protective suit, and \$8.50 for a respirator with cartridge. The results were as follows:

Table 2 Browns Ferry Candidate Areas

No.	Room Name	Background	Inspect Entry	No. Workers	Temperature
1.	Steam Jet Air Ejector	500 mR/hr	1/week	1	120°F
2.	Chemical Waste Tank	30 mR/hr	daily	2	90°F
3.	Spent Resin Tank (EL 546)	450 mR/hr	1/month	2	90°F
4.	Clean Pump	375 mR/hr	1/month	3	90°F
5.	Moisture Separator	2 R/hr*	1/month	4	140°F
6.	HP Feedwater Heater	10 mR/hr	2/month	3	160°F
7.	LP Feedwater Heater	10 mR/hr	2/month	3	160°F
8.	Moisture Separator Drain Pump	200 mR/hr	1/month	3	100°F
9.	Feedwater Pump	125 mR/hr	1/week	2	100°F
10.	Drywell (Basement)	50 mR/hr	outage	5	90°F
11.	Drywell (Rod Gallery)	125 mR/hr	outage	5	90°F
12.	Drywell (Upper Elev.)	40 mR/hr	outage	5	90°F
13.	Condensate & Waste Sludge	35 mR/hr	1/week	1	90°F
14.	RHR Heat Exchanger	700 mR/hr	outage	4-6	90°F
15.	Waste Backwash Receiver Tank	450 mR/hr	1/week	2	90°F
16.	Top of Torus	50 mR/hr	1/month	2	90°F
17.	Moisture Separator EHC Valves	2 R/hr	3/year	4	130°F
18.	Moisture Separator Tank Room	100 mR/hr	3/month	3	100°F
19.	Clean Backwash Receiver Tank	40 R/hr	1/quarter	3	90°F
20.	Waste Packaging	200 mR/hr-15 R/hr	daily	2	90°F
21.	Smoke Detectors (All Rooms)	Variable	2/year (each detector)	3	Variable

*During reactor operation

Source: Browns Ferry ALARA Coordinator

Area	Cost to Implement	Annual Plant Savings
• PWR Upper Ice Condenser	\$ 78,200	\$ 41,700
• PWR Seal Table	\$ 13,000	\$ 93,900
• PWR Reactor Vessel Head	\$ 19,000	\$ 18,100
• PWR Keyway	\$ 32,400	\$ 2,000
• BWR Smoke Detector	\$ 20,000	\$ 9,600 (Minimum)
• BWR Moisture Separator and Feedwater Pump Rooms	\$150,000	\$104,800
• BWR Feedwater Heater Rooms	\$ 23,400	\$ 16,800

A major conclusion is that robotic surveillance and inspection devices can be retrofitted into existing nuclear power plants and will reduce both radiation exposure to workers and plant operating costs. Robotic implementation costs can be minimized by using commercially-available robotic components, not requiring plant modifications and using the robots in a number of different areas. The benefit analysis should consider all costs associated with suited personnel entry into a radiation area. It is important to recognize that benefits will differ significantly between plants, and cost/benefit analyses should be performed on an individual plant basis.

SURBOT DEMONSTRATION

The three moisture separator rooms (Figure 2) and nine feedwater pump rooms at the Browns Ferry Plant were identified in the Phase I study as prime candidates for the use of a roving surveillance robot. Each separator room is entered an average of sixteen times per year by a four-worker team. Each pump room is entered once per week for radiation measurements and once per shift to read gauges and to make a visual inspection. Conditions within the separator rooms are 160°F ambient temperature and 2 R/hr during reactor operation. The pump rooms have a 100°F temperature and 25-125 mR/hr background. Full C-zone clothing is required for most entries. It is estimated that the use of a surveillance robot in these rooms could produce the following annual benefits for the Browns Ferry Plant.

- Reduced HP technician labor = 380 man-hours/year = \$7,600
- Reduced operator labor = 500 man-hours/year = \$10,000
- Reduced worker exposure = 70 man-rems/year = \$70,000
- Reduced C-zone clothing = 1500 sets/year = \$15,000
- Improved worker safety

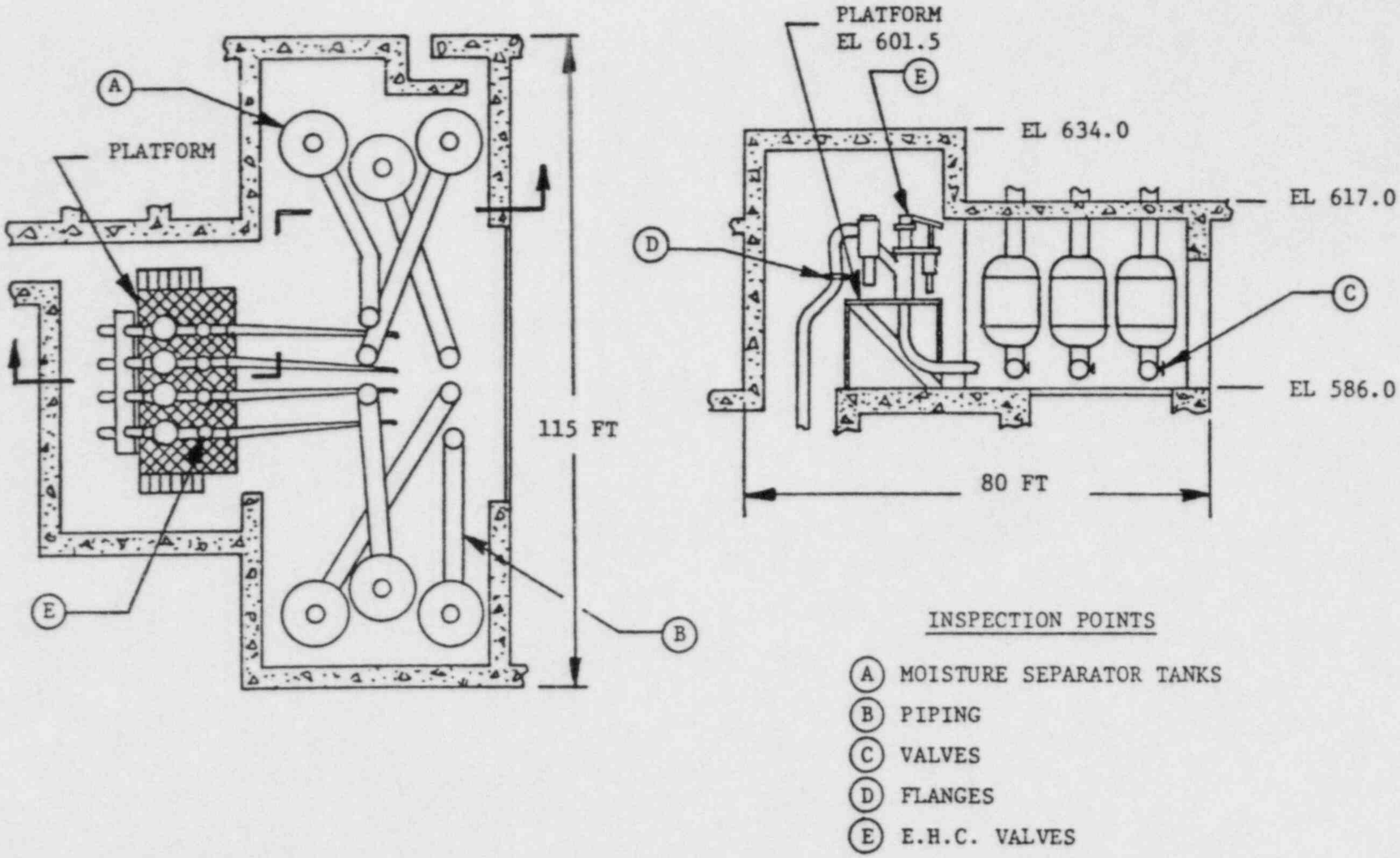

Figure 2 Moisture Separator Room.

The Phase II project includes the design and construction of a surveillance robot, called SURBOT, following the basic concept shown on Figure 3. The SURBOT is scheduled to begin hot testing at the Browns Ferry Nuclear Plant in January 1986.

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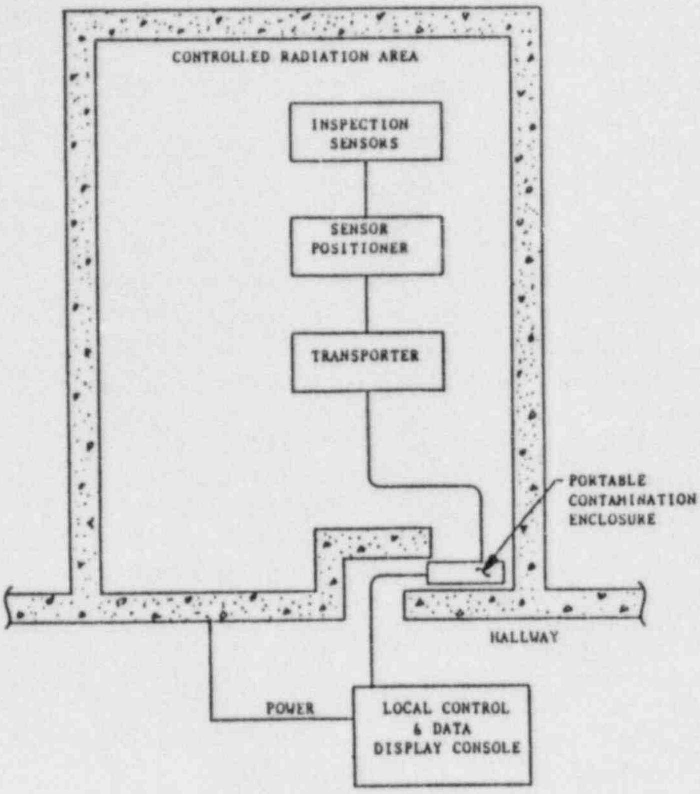

Robot Guidelines

- Use commercially-available components
- Minimum plant modifications
- Cabling for power/signal
- Automatic or operator controlled
- Full height coverage of area
- Hard copy data recording
- Usable in different radiation areas
- Positive contamination control
- No worker suit-up to operate
- High reliability
- Failure recovery provision

Inspection Requirements

- Detect steam/water leaks
- Inspect pipe snubbers
- Verify valve positions
- Read gauges
- Detect loosened parts
- Detect electrical arcing
- Verify security locks
- General noise monitoring
- Measure radiation level of components
- Air sampling
- Contamination smearing
- Liquid spillage sampling
- Observe maintenance workers

Figure 3 Surveillance Robot Concept.

IMPACTS OF DECONTAMINATION OF LWRs ON SOLIDIFICATION AND WASTE DISPOSAL

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INTRODUCTION

The nuclear industry is actively considering the potential advantages of primary system decontamination to ensure the safe operation of light water reactors (LWR). The Nuclear Regulatory Commission (NRC) is responsible for insuring the public health and safety, and will therefore require a careful evaluation of different decontamination processes and the unique wastes they produce. The areas which are being addressed in this work to aid the NRC in their evaluation of the effectiveness and safety of chemical decontamination processes are: the type, volume and toxicity level (radiotoxicity as well as chemical toxicity) of the radwaste streams generated by the decontamination as well as their subsequent management at the plant and at the disposal site.

The objectives of the program at Brookhaven National Laboratory (BNL) are to identify the information and conduct the tests necessary to aid the NRC in making regulatory decisions on the waste disposal aspects of chemical decontamination processes. Because of the large amounts of chelates or complexing agents required for a full system decontamination, it is desirable to determine if there are methods which would convert these reagents to more acceptable forms prior to disposal. In particular, this work has focused on direct solidification of decontamination wastes and processes for converting decontamination wastes to more innocuous forms.

SOLIDIFICATION OF SIMULATED DECONTAMINATION RESIN WASTES

The waste generated by the chemical decontamination of a light water reactor will be disposed of in a shallow land burial site. Disposal of these wastes will come under the minimum requirements established in the rule "Land Disposal of Radioactive Waste," 10 CFR Part 61, any future guidance given by the NRC and through site specific criteria. A laboratory evaluation of methods for solidifying decontamination wastes was performed in order to assess whether the soldified wastes will meet the applicable criteria. The simulated decontamination resin waste composites were examined for the presence of free liquid, tested for mechanical strength and tested for their ability to withstand immersion in water. Further, mechanical strength tests were performed after the water immersion testing.

Simulated dilute decontamination resin wastes were solidified in Portland I cement and vinyl ester-styrene (VES). The following reagents were used in the testing: ethylenediaminetetraacetic acid (EDTA), oxalic acid (OA), citric acid (CA), EOC (an equimolar mixture of EDTA, OA and CA), picolinic acid (PA), formic acid (FA), simulated LOMI reagent (an equimolar mixture of PA and FA was used) and LND 101A (a proprietary reagent supplied by London Nuclear Limited). Two anion-exchange type resins were used: IRN-78 (Rohm and Haas), a polystyrene strong base anion exchange resin in the ^{-}OH form and IONAC A-365 (Sybron), a polyacrylic weak based anion exchange resin with exchange groups in the free base and ^{-}OH forms. Samples made with mixed bed resin had IRN-77 (Rohm and Haas) as the cation exchange resin used in the H^+ form. Enough IRN-77 was used to produce a weight ratio of two parts anion exchanger to one part cation exchanger. The anion exchange resins were equilibrated with an amount of acid that would exchange with 50% of the available sites.

Portland I cement was selected for this work based on the results of a series of scoping experiments.[1-3] The resin waste slurry was pretreated, prior to solidification in cement, with sodium hydroxide to increase the pH to approximately 12*. Anion resin wastes solidified in VES were adjusted to pH ≈9.5 with hydrochloric acid if necessary whereas mixed bed resin samples were solidified without pretreatment. The laboratory scale waste forms had either a nominal 2-in. diameter by 4-in. height or a 2.6-in. diameter by 3.1-in. height. Details of the solidifications are given in Reference 1 and references therein.

Simulated Resin Wastes Solidified in Cement

In all waste forms except those containing citric acid, the forms cured to free standing monoliths with no free liquid within 28 days. Cement solidified mixed bed resins containing citric acid cured to a hard set after ≈90 days. Citric acid/anion resin composites had not cured after ≈90 days. The compressive strengths of cement solidified wastes were measured by the ASTM-C39-80 test method. All waste composites exhibited compressive strengths within a range of 2100 to 3400 psi.[3] This is well in excess of the 50 psi minimum recommended in the NRC Technical Position on Waste Form.

Five waste forms of each waste type were immersed in one liter of deionized water (DIW) to test the ability of the composites to maintain their physical integrity during continued exposure to water. During the 90-day immersion tests of mixed bed resin wastes solidified in cement, three of five forms containing LND-101A disintegrated. Mixed bed resin waste containing Na_2EDTA also exhibited a cracked or scaled surface although no flaking was observed. Cement composites of picolinic acid on polystyrene mixed bed resin showed a cracked pattern on the upper third of the waste form. All citric acid/mixed bed resin/cement composites cracked during 90 days of immersion. All of the forms that were suitable for compression testing had compressive strengths greater than 50 psi after immersion in water.[4]

Following 90 days of immersion, waste composites containing LND-101A on anion resins had cracked. Fractures were also evident on anion resin composites containing EDTA-oxalic acid-citric acid (EOC) and on composites containing Na_2EDTA. All waste composites that maintained integrity after water immersion had compressive strengths greater than 50 psi.[5]

*Anion resins containing LOMI reagent were treated with hydrochloric acid to decrease the pH to about 5.5 in order to slow the set time and allow sufficient mixing of the composite.

Simulated Resin Wastes Solidified in VES

The mechanical strengths of the simulated decontamination resin waste/VES composites were measured according to the ASTM C39-80 test method with the exception that the forms were measured for compressive strength at 10% deformation. All of the mixed bed resin composites had compressive strengths well in excess of the 50 psi recommended in the Technical Position on Waste Form. The values ranged between 1100-1300 psi with the exception of the controls (10% deformation strength ≈950 psi).[6] No liquid was observed seeping out of these composites during deformation testing. Immersion in DIW was performed using one 2.6 in. x 3.1 in. composite of each decontamination waste type of mixed bed resins solidified in VES. No sample deterioration or expansion was observed following 90 days of immersion.

The compressive strengths at 10% deformation of simulated anion resin-VES composites were in the range of 1200 to 1500 psi. Samples containing simulated wastes in general have lower compressive strengths than the control sample.[7]

During the compression tests of the anion resin composites liquid was observed seeping from the surface of specimens containing IRN-78 resins with either picolinic acid, oxalic acid, citric acid, EOC or LND-101A but not the control specimens.[7] Both the IONAC control and LOMI specimens released liquid under compression. Measurements of liquid pH were made on some samples using narrow range pH paper. Although the accuracy of the measurements may be poor the liquid released from IONAC A-365 control appeared neutral whereas some of the picolinic acid/IRN-78 forms released liquid which appeared to be acidic. The presence of the organic acids and the resin type in the VES forms may influence the behavior of the form. Therefore the presence of decontamination reagents in a waste stream should be considered when establishing a solidification process using VES.

Five samples of each VES composite containing reagent on anion resins were immersed in one liter of DIW. After 90 days of immersion, no sample deterioration or swelling was observed.

Free liquid was observed on removal of the VES waste forms from the containers. Two different free liquid generation tests were performed on simulated mixed bed resin wastes/VES composites. In conjunction with these tests, two different measuring techniques were employed. The initial measurement was concerned with what is defined as drainable liquids (ANSI/ANS-55.1-1979), but in this study the term pourable liquids was substituted. If a sufficient quantity of unbound liquid remained in the polyethylene container following removal of the composite, the container was then inverted and the liquid contents poured off and weighed. (No pourable free liquid measurements were performed on 2.6 in. x 3.1 in. composites.) The second measurement was performed using an absorbent tissue. Any weep water located on the exterior of the form or residual water on the inside container walls was absorbed with a tissue and quantified by weighing.

The second free liquid test involved repeated determinations of any moisture generated following the initial drying of the container and composite. (Only the 2 in. x 4 in. composites were tested using this procedure.) Measurements on one sample from each acid type and the control were determined. The free liquid performance of these composites (based on the ANSI/ANS-55.1-1979 criterion, although pourable liquids are not included) was calculated and is summarized in Table 1.

Free liquid measurements were made on anion resin-VES composites. These data are given in Table 2 and represent a single determination (five replicates) of each waste type of the "pourable" and tissue sorbed liquid. No measurements were made of free liquid generation with time although a separate set of composites were measured approximately one week after the initial determinations. The amount of free liquid (pourable and tissue sorbed) was equivalent to that measured initially with the error given.

Based on single measurements of the free liquid, most waste composites are within the 0.5% limit for solidified waste as recommended in the Technical Position on Waste Form. The potential exceptions are forms containing citric acid/mixed bed resin waste, EOC/polystyrene anion resin waste and simulated LOMI wastes on polyacrylic resins. In the case of the LOMI/polyacrylic resin composites, the percent free liquid by volume approaches the limit for Class A wastes. If the standard had been applied to those mixed bed composite samples measured for recurring free liquid generation, they all would have been in excess of the 0.5% for solidified wastes.

INCINERATION OF SIMULATED DECONTAMINATION WASTES

There are several different incinerator designs which range from a starved or controlled air incinerator, to molten glass and salts incinerators. (For a brief description see Reference 9.) It is not the intent of this program to evaluate all different types of incinerators but to assess whether combustion of decontamination wastes is a viable means of destroying or degrading these wastes prior to final packaging for disposal. A second objective is to determine, where possible, how process parameters (e.g., temperature) may affect the overall applicability of the process.

A tube furnace was set up to study the degradation of simulated decontamination wastes by incineration. Details of the experimental apparatus and analytical procedures are described elsewhere.[1,6] Several modifications were made on the laboratory-scale incinerator to increase the efficiency of the combustion process. A pure oxygen feed was used as the oxidizing gas. When compressed air was used, excessive amounts of ash and soot were observed. An afterburner was added to further oxidize the combustion gas and a CuO catalyst was used to insure the oxidation of reduced carbon species (e.g. CO) to CO_2. The length of the main chamber and the afterburner limit, to some extent, the amount of sample that can be combusted.

Table 1. Percentage free liquid. Simulated mixed bed resin decontamination wastes in VES.

Acid	Composite Size: 2.6 in. x 3.1 in.[a] One Measurement	Composite Size: Nominal 2 in. x 4 in.[b] One Measurement[d]	Composite Size: Nominal 2 in. x 4 in.[b] Repeated Measurement[c]
Control	--	0.32	0.79
Oxalic	0.16	0.25	0.76
Picolinic	0.09	0.35	0.90
Formic	0.07	0.25	0.78
Disodium EDTA	0.44	0.40	0.54
Citric	0.47	0.51	1.06

[a]Calculations were performed using a composite volume of 284.93 cm^3.
[b]Calculations were performed using a composite volume of 158.10 cm^3.
[c]Measurements were made over a period[1] of 35-days.
[d]A total of pourable and sorbed liquid.

Table 2. Percentage of free liquid. Simulated anion resin decontamination wastes[a] in VES.

Acid/Resin	% Pourable[b] Liquid by Volume	% Total[b] Liquid by Volume
EDTA/PS[c]	0.1	0.2
PA/PS	0.3	0.3
EOC/PS	0.4	0.5
OA/PS	0.3	0.4
FA/PS	0.2	0.3
CA/PS	0.2	0.3
LND-101A/PS	0.2	0.3
LOMI/PA[d]	0.9	1.
C-PS[c]	0.2	0.2
C-PA[d]	0.1	0.2

[a]Values based on 5 replicate measurements.
[b]Average volume assumed to be 165 $cm^3 \pm 2.0$ cm^3 from random measurements of diameter and height of several samples.
[c]PS is Amberlite IRN-78 anion exchange resin; C-PS is a control sample containing resin only.
[d]PA is IONAC-365A anion exchange resin; C-PA is a control sample containing resin only.

Destruction of the materials was monitored by a carbon mass balance. The off gas of the incinerator was passed through a sodium hydroxide gas scrubber to trap CO_2. The quantity of carbon from CO_2, determined by titration of carbonate-bicarbonate, was related to the initial carbon present in the sample. Organic acids, organic ion-exchange resins, and organic ion-exchange resins equilibrated with organic reagents were incinerated.

A summary of the incineration data is given in Table 3. In solid samples of organic reagents and the proprietary reagent LND-101A, 80% and greater of the available carbon could be accounted for during incineration of these samples. The carbon recovered during incineration of the resin samples appears also to be in the range of 80% or greater. However, there is larger uncertainty in this value due to uncertainty in the carbon content and water content of the resins. Species other than CO_2, CO and CH_4 have been identified during the incineration of EDTA but in trace amounts.[8] These include HCN and NO. Fourier Transform Infra-Red spectra of gas samples taken before the copper oxide catalyst indicate the presence of trace quantities of N_2O and HNO_3 during combustion of EOC and N_2O, HNO_3, NO_2 and HCN during combustion of IRN-78. However, the dominant species in all instances was CO_2.

The results for the incinerations of resins equilibrated with decontamination reagents are comparable with those for the resins and acids alone. In most cases greater than 90% of the available carbon was accounted for in the NaOH gas traps. If the estimates of available carbon in the acid/resin samples tested were low then the reported percent carbon recovery for these samples may be high. However, the reproducibility of the tests demonstrates that the process was consistently effective at destroying the reagents and resins. No clear explanation can be given for the 70% carbon recovery observed for samples EOC/IRN-78 (2) and PA/IONAC A-365 (2).

Some parameters are waste specific. For example, the temperature of the main chamber normally has to be adjusted for the sample being combusted. While a temperature of 700°C is adequate for a solid sample of EDTA, citric acid burns too quickly at this temperature. This results in ash and char being deposited on the main chamber and into the afterburner. With damp resins (resins in the as-received form) in a pure O_2 stream, the oxidation could be more easily controlled. This may indicate that a change in combustion temperature is required for dried resins.

If an incinerator is broken down into three subsystems (feed system, the incinerator, and the off-gas system) the following comments may be applicable to large-scale processing. We have found that combustion proceeds more completely if the sample is introduced into a hot combustion chamber. When samples were placed in a cold furnace and the temperature increased, char or soot formed. (Once the furnace is at temperature, these can be burned off within the hot zone.) In the incinerator itself we have used both compressed air and oxygen as oxidizing mediums. When compressed air is used, particularly with resin samples, large amounts of soot and smoke are formed. With compressed air or a depleted oxygen stream, pyrolysis can occur. The choice of oxidizing medium would then affect the rest of the incinerator design. Depending on the sample size, the soot and smoke can be transported as far as the gas trapping system. This is due, in part, to the larger gas flow rates required with a

Table 3. Summary of incineration tests -- titration results only.

Sample		Available Carbon[a] (g)	Trapped Carbon[b] (g)	Trapped Carbon (%)
EDTA	1	0.8215	0.717 ±0.01	87.3
	2	0.8199	0.821 ±0.01	100.2
	3	0.8512	0.861 ±0.01	101.2
	4	0.8299	0.839 ±0.01	101.1
	5	0.9188	0.858 ±0.02	93.4
	6	0.8207	0.646 ±0.01	78.7
	7	0.8499	0.753 ±0.01	88.6 (92)
	8	0.8258	0.731 ±0.02	88.5 (93)
Citric acid	1	0.8569	0.827 ±0.02	96.5
	2	0.8582	0.730 ±0.02	85.0
	3	0.8481	0.743 ±0.01	87.6
	4	0.8726	0.903 ±0.01	103.5
	5	0.8582	0.835 ±0.01	97.4 (99)
Picolinic acid	1	0.8996	0.834 ±0.01	92.8
	2	0.8975	0.8185±0.01	91.2 (93)
	3	0.8959	0.8678±0.001	96.9 (99)
EOC	1	0.8092	0.625 ±0.01	77.2 (84)
	2[c]	0.8170	0.3888±0.01	47.6 (57)
	3	0.8093	0.6380±0.01	78.8 (85)
LND-101A	1	0.7822	0.735 ±0.01	94.0
	2	0.7682	0.588 ±0.01	76.5 (84)
	3	0.7755	0.534 ±0.01	68.8 (76)
IRN-78	1[d]	3.50 ±0.02	1.952 ±0.05	84.3
	2[d]	3.40 ±0.02	3.597 ±0.05	105.7
	3[d]	3.38 ±0.02	3.275 ±0.01	96.9
	4	0.803±0.004	0.6796±0.015	84.6 (89)
	5[e]	0.800±0.004	0.439 ±0.03	54.9 (58)
	6	0.834±0.004	0.7734±0.01	92.7 (94)
IONAC-365	1	0.789±0.01	0.739 ±0.05	93.7 (96)
	2	0.821±0.01	0.7271±0.09	88.6 (90)
	3	0.817±0.01	0.7049	86.3
EDTA/IRN-78	1	0.658±0.007	0.635 ±0.001	96 (97)
EOC/IRN-78[f]	1	0.879±0.009	0.881 ±0.003	100 (101)
	2	0.914±0.008	0.696 ±0.01	76 (77)
PA/IONAC A-365	1	0.79 ±0.02	0.710 ±0.003	90 (91)
	2	0.83 ±0.02	0.58 ±0.01	70 (71)
	3	0.79 ±0.02	0.663 ±0.004	84 (85)
LND-101A/IRN-78	1	0.83 ±0.01	0.77 ±0.02	93 (94)
	2	0.87 ±0.01	0.858 ±0.01	99 (100)
	3	0.79 ±0.01	0.71 ±0.01	90 (91)

[a] Available carbon based on molecular weight of reagents. Citric acid is in a monohydrate form. EOC (EDTA-oxalic acid-citric acid) is an equimolar mixture of these acids. Available carbon for the anion resins is based on 33% for as-received IRN-78 and 31% by weight for as-received IONAC-365 resins (BNL-NUREG-33873, 1983)

[b] The amount of trapped carbon is determined from the equivalents of HCO3- measured by titration from pH = 8.3 to pH = 4.5.

[c] Leak was detected in the gas sampling system.

[d] These data were reported prior to resin analysis. The data have been corrected to assume 33% by weight carbon for as-received resins.

[e] Leak suspected in gas sampling system.

[f] Molar ratio of EDTA:OA:CA was: 1:2:2 for sample 1 and 3:2:2 for sample 2.

compressed air feed stream. Therefore, the size of the sample, the feed stream, and the length of the combustion zone are all important process parameters. These three factors all affect the residence time in the hot zones.

For the gas handling systems, we have chosen to use an afterburner and a CuO catalyst to insure complete oxidation before trapping with caustic solution. An independent air feed to the afterburner may eliminate some of the soot and smoke evolved with a compressed air feed. It was found that the trapping efficiency of the scrub solutions depended on the sample size and the configuration of the traps. If all off-gases are to be trapped by this type of system, these solutions may require frequent replenishment since the trapping efficiency decreases rapidly as the solution pH drops. The combustion system we have used is all glass and no statement can be made about potential corrosion problems. Corrosion is known, however, to be a prime consideration with incinerators.[9] In addition, a scrub solution such as the one used here (1 N NaOH) may cause corrosion in the off-gas system if the materials chosen are susceptible to attack by alkaline solutions.

No solid wastes remained in the incinerator as expected in processing actual radwaste. This is primarily because pure acids or acid resin samples were combusted and conditions were optimized for complete combustion. However, the off-gas scrub solutions will contain large amounts of Na_2CO_3 and possibly NaOH or $NaHCO_3$, depending on the pH. The acid scrubber may also contain nitrates, sulfates and chlorides, depending on the type of wastes. These may require further management prior to disposal.

In conclusion, resins, acids and acid/resin combinations can be successfully destroyed by incineration. Oxygen as a carrier gas and a high after burner temperature both promote a high degree of oxidation of these carbon containing reagents to CO_2. Process temperature control was important to minimize soot or maximize oxidation of the samples.

CHEMICAL DIGESTION: SULFURIC ACID AND HYDROGEN PEROXIDE

While many chemical digestion systems have the potential to degrade or destroy chemical decontamination wastes, acid digestion as a process has been developed to a pilot plant scale for use with low level radioactive wastes and was therefore selected for study in the BNL program. After scoping studies with both HNO_3 and H_2O_2 as the secondary oxidant, H_2O_2 was selected for its ease of handling and its apparent better oxidation capabilities. No studies have been done of the effect of catalysts and only minor variations in process parameters have been studied. The primary goal was to determine if chemical digestion, specifically acid digestion, is effective for processing simulated chemical decontamination wastes.

The acid digestion unit has been described in detail elsewhere.[1,8] The chemical system employed consisted of a hot sulfuric acid bath with hydrogen peroxide as the secondary oxidant and oxygen as a carrier gas. As for the incineration tests, the acid digestion off gas passed through a CO_2 trap system and the fraction of destruction was determined by carbon mass balance.

The data listed in Table 4 indicate that more than 70% of the available carbon of the reagents or the resin can be accounted for if the acid digestion process conditions are optimized. Similar results are achievable in the acid digestion of resin samples equilibrated with candidate decontamination reagents. Equipment failure during the acid digestion of one sample of LND-101A/IRN-78 caused one low value (36%) for carbon recovery. The reproducibility of the carbon recovery values in general demonstrated that the acid digestion process was consistently effective at destroying the resins and reagents. However, the accuracy of the results is dependent on the accuracy of the calculation of carbon in the original acid/resin samples. For example, if the amount of carbon in the acid/resin samples was underestimated then the percent carbon recovery reported may be too high.

The digestion of EOC/IRN-78 (2) indicates that this acid digestion system can handle samples having 2 grams of available carbon. However, it was necessary to prolong the reaction time. A larger percent carbon recovery may have been achieved had more than 52 mℓ of H_2O_2 been used. The result indicates that the amount of H_2O_2 used for the other runs was in excess. No effort was made to determine the minimum quantity of H_2O_2 needed for digestion of samples containing 0.5 g of carbon.

The results from the digestion LND-101A (Table 5), indicate that a combination of smaller sample size, longer reaction time, large amounts of H_2O_2 and the addition of a CuO catalyst all enhance the destruction of the reagent and reduce the amount of by-products (e.g. CO) as determined by the titration data.

Table 5. Summary of LND-101A digestion data.

	Sample (g)	Gas Flow (mℓ/min)	Temp.	H_2O_2 Added (mℓ)	Reaction Time	Percent Recovery
LND-101A (1)	2.0000	30	≈250	20.1	135	62
LND-101A (2)	2.3748	30	≈250	7.0	21	59
LND-101A (3)	1.5022	30	≈250	53.8	70	92

The optimum conditions for each reagent and for resins have not been determined although several tests have indicated those parameters that are most important to the process. These include amount and rate at which the secondary oxidant is added, sample size, temperature and for best efficiency, a secondary system for fully oxidizing other carbon containing gases (e.g. CO, CH_4) and volatiles (e.g. acetone). The process has been found to be more efficient if the secondary oxidant is added continuously and if the digestion is allowed to proceed beyond the time when the acid bath appears clear. This may, in part, be due to the greater difficulty in oxidizing lower molecular weight by-products. If the temperature of the digestion is decreased, longer reaction times and more peroxide may be required, but process control may be

Table 4. Summary of acid digestion tests -- titration data only.

Sample		Available Carbon (g)	Trapped Carbon (g)	Trapped Carbon (%)
EDTA	1[a]	0.8601	0.43 ±0.03	50
	2[a,c]	0.9081	0.29 ±0.01	32 (37)[b]
Citric acid	1[a]	0.8573	0.60 ±0.01	70 (76)
	2	0.8575	0.70 ±0.01	82 (88)
Picolinic acid	1[d]	0.5002	0.17 ±0.01	34 (35)
	2	0.5003	0.38 ±0.01	77 (79)
	3	0.5026	0.48 ±0.01	96 (97)
EOC	1	0.5348	0.43 ±0.01	80 (83)
	2	0.5313	0.33 ±0.01	62 (65)
	3	0.5332	0.33 ±0.01	63 (65)
LND-101A	1[a]	0.6680	0.42 ±0.01	62 (64)
	2[a]	0.7932	0.47 ±0.03	59 (63)
	3	0.5017	0.46 ±0.01	92 (98)
IRN-78	1	0.57 ±0.01	0.48 ±0.01	84 (90)
	2	0.57	0.502±0.001	88 (89)
IONAC-365	1	0.54 ±0.01	0.44 ±0.01	82 (88)
	2	0.63 ±0.02	0.48 ±0.01	76 (81)
	3	0.53 ±0.01	0.45 ±0.01	87 (92)
EOC/IRN-78[f]	1	0.547±0.006	0.515±0.009	94 (99)
	2	2.191±0.03	1.73 ±0.06	79 (81)
	3	0.728±0.005	0.580±0.007	80 (82)
PA/IONAC A-365	1	0.53 ±0.01	0.371±0.006	70 (73)
	2	0.57 ±0.01	0.46 ±0.01	80 (83)
	3	0.48 ±0.01	0.405±0.009	84 (87)
LND-101A/IRN-78	1	0.540±0.008	0.197±0.004	36 (39)[e]
	2	0.643±0.009	0.59 ±0.02	91 (94)
	3	0.577±0.008	0.49 ±0.02	85 (87)

[a]These samples were digested prior to the addition of a CuO catalyst. Further modifications of the digestor resulted in more CO2 trapped in later digestions.
[b]Value in parentheses gives the percent destruction based on the titration data and the gas sample analysis.
[c]Acid bath broke during the digestion.
[d]It is thought that the digestion was stopped prior to completion.
[e]Apparatus broke during run. It is suspected that CO2 escaped from the system.
[f]Molar ratio of EDTA:OA:CA was: 1:2:2 for sample 1, 1:5:5 for sample 2, and 1:1:1 for sample 3.

easier. (The reaction time in this instance could, in principle, be shortened by the use of a suitable catalyst.) In some initial tests with HNO_3 as the secondary oxidant, the process was more difficult to control than with hydrogen peroxide. However, in the work conducted in this program no effort has been expended to investigate other oxidants or the potential usefulness of a catalyst. The tests conducted focus primarily on the ability of the process to convert the wastes to more innocuous forms. No mechanistic studies have been done to elucidate those areas where changes in or additions of reactants would optimize the process.

In conclusion, resins, acids and acid/resin samples can be successfully processed by acid digestion. Oxygen carrier gas and excess H_2O_2 were used to promote oxidation. However, a CuO catalyst was necessary to convert CO to CO_2. Under the conditions studied acid digestion appeared to be less efficient than incineration. This was evidenced by the larger amounts of methane and CO which passed through the CuO catalyst and CO_2 traps. No efforts were made to examine the behavior of samples containing metal ions and/or cation exchange resins which may be present in actual decontamination wastes. The iron-catalyzed oxidation of ion-exchange resins in hydrogen peroxide (alone) was reported by Hawkings and others.[10] In that study the digestion was effective at temperatures of ≈100°C but was limited to treating cation exchange resins having a polystyrene-DVB matrix. Although the process was found more efficient with iron present the authors warn that too large an excess of iron can result in a rather vigorous reaction. Since iron ions are present in decontamination wastes they may enhance the amount of degradation achieved and also affect the control of the acid digestion process.

SUMMARY

From scoping studies it was determined that Portland I cement was potentially a better binder than Portland II, III and masonry cement.[1-3] Some pretreatment of the waste prior to solidification also appears to increase the resistance of the waste composite to continued immersion in water. However, the tests conducted in this program point to the need for a waste-specific determination of the acceptability of these wastes for disposal, i.e., the ability of the waste composite to meet the applicable NRC criteria and site specific criteria. Work at BNL has indicated the following: wastes containing large amounts of citric acid require a long cure period before forming a hard free-standing monolith; composites containing polyacrylic resins equilibrated with picolinic acid require a different pre-treatment (i.e., addition of acid) than those wastes containing polystyrene resins; even with pretreatment certain waste types appear to degrade and others fail after long periods of immersion in water. All of this work indicates that it may be necessary to institute a process control program to determine what compositions will be acceptable for the waste being solidified and the range of conditions (e.g. waste loading, cement-to-water ratio, additive concentration) under which the waste can be solidified to an acceptable product.

In general, solidification of simulated decontamination resin wastes in a vinyl ester-styrene binder results in a free standing monolith. The waste appears to be homogeneously distributed in the waste/binder composite, to have a compressive strength at 10% deformation of greater than 1000 psi, and to be able to withstand immersion in water.

Those areas that may require further investigation on how they may affect in-plant process control and the behavior of the waste form when disposed of in a shallow land burial site are the presence of and development of free liquid and the apparent increase in the surface porosity of some acid/resin waste forms.

Work conducted at BNL on the incineration and acid digestion of simulated decontamination resin wastes indicated that both processes can, in principle, be very effective for degrading these wastes. Under the conditions studied incineration appeared to be slightly more effective than acid digestion. However, it should be noted that the work conducted at BNL was limited to a single combustion unit, a single chemical digestion system and three simulated waste streams: reagents alone, anion resins alone, and anion resins equilibrated with candidate chemical reagents. Only small variations in process parameters were considered. While the data indicate the applicability of these processes to these types of wastes, it is recommended that any process being considered for use should be tested with the appropriate waste stream or a simulation of the waste stream to insure that conditions are adequate for processing and degradation of the wastes.

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Effectiveness and Safety Aspects of Selected Decontamination Methods for LWRs

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Abstract

The Effectiveness and Safety Aspects of Selected Decontamination Methods program objective is to obtain information on chemical decontamination methods that the Nuclear Regulatory Commission (NRC) might be expected to review in the near future. General topic areas are the effectiveness in reducing occupational dose to plant individuals, the volume of radioactive waste generated as a result of decontamination efforts, and plant information that would assist the NRC in their review process. To date, our evaluation has concentrated on decontamination factors (DFs) obtained, methods used in determining DFs, man-rems savings, waste forms generated, man-rem expended in performing the decontamination, and an evaluation of the overall operation (i.e. problems and lessons learned.) This paper presents the preliminary results of these evaluations.

Introduction

The increased utilization of in-situ chemical decontamination of primary coolant systems and associated equipment has raised questions centered around the effectiveness of decontamination efforts and ease of system decontamination. To assist the NRC personnel in finding answers to these and related questions a research program "Effectiveness and Safety Aspects of Selected Decontamination Methods" was funded. This program was started in the fall of 1983 with these objectives:

- study current decontamination criteria, techniques, data, results, and problems;
- make confirmatory and/or supplemental measurements;
- evaluate the effectiveness of the studied decontamination methods to lower potential or actual personnel exposures;
- evaluate the current techniques used for determining decontamination factors (DFs).

To date, research personnel have visited seven nuclear power reactors to observe in-situ chemical decontaminations performed on primary system components. Six of these chemical decontaminations were performed on recirculation system piping in boiling water reactors (BWRs) prior to pipe replacement or weld inspection. One decontamination was a steam generator cleaning prior to tube plugging operations in a pressurized water reactor (PWR). Also, several BWRs performed chemical cleaning in the reactor water cleanup system (RWCS). The chemical decontamination processes that have been observed are CAN-DECON[1], LOMI[2], and DOW NS-1[3]. Confirmatory and supplemental measurements have been performed at four of the BWRs. The confirmatory measurements were made to evaluate the licensee's measurements. The supplemental measurements were made to assist EG&G personnel in evaluating techniques for determining DFs and specific radionuclide behavior.

Methods

Program personnel have traveled to seven sites to observe the selected decontamination process used to remove the oxide films to which radionuclides were bound. Discussions with facility and vendor personnel, and later confirmatory and supplemental measurements, were conducted at each facility. The discussions addressed several subject areas:

- criteria for choosing to use chemical decontamination;
- projections of man-rem savings and how they were achieved;
- problems associated with decontamination;
- personnel exposure;
- methods used to determine the end point of the decontamination;
- measurements and technique used to evaluate DF;
- any prior decontamination;
- waste handling and volume of waste;
- other related techniques used to reduce total man-rem.

The confirmatory and supplemental measurements that were made were gross measurements with an Eberline E-530N/HP 220 A GM probe, thermoluminescent dosimeters (TLDs), and spectral measurements with an Ortec CPD-1 intrinsic germanium detector with a tungsten shield. These measurements were made both pre and post decontamination to determine how DF varies with measurement techniques. Measurements using an Ortec CPD-1 system were performed at one of the facilities pre and post decontamination and at two other facilities post decontamination. The latter measurements were made as part of a recontamination study.

Discussion

For uniform understanding both chemical decontamination and decontamination factor (DF) will be defined by the authors as:

> Chemical decontamination: the removal of a system radionuclide inventory using chemical reagents to dissolve the oxide films in which the radionuclide is incorporated as an integral part of the oxide structure or as an impurity.
>
> Decontamination Factor: a numerical representation of the effectiveness of a decontamination process and is calculated as follows:

$$DF = \frac{\text{pre-decontamination measurement}}{\text{post-decontamination measurement}}$$

As the study progressed it was soon apparent that the nuclear industry used two different types of DFs, which we will term man-rem DF and decontamination DF. We define these as:

> man-rem DF: the decrease in general area radiation body fields realized by decontaminating various systems and components.
>
> Decontamination DF: the decrease in surface radionuclide inventory realized by decontaminating various systems or components, which decrease is usually determined by contact dose rate measurements.

To further complicate determining DFs it has been observed that several different techniques are being used to determine an average DF value. These methods are arithmetic mean, and weighted harmonic mean. The equations or method used for determining these DFs are:

Arithmetic Mean $\overline{X} = \frac{1}{n} \sum_{i=1}^{n} DF_i$

where: n = number of sample points
DF_i = decontamination DF at each sample point

Weighted Harmonic $\overline{X}_H = \frac{\sum_{i=1}^{n} \dot{D}P_i}{\sum_{i=i}^{n} \dot{D}F_i}$

where: $\dot{D}P_i$ = dose rate prior to the decon at point i
$\dot{D}F_i$ = dose rate after the decon at point i

Another method used by EG&G personnel was a median DF value which is defined as:

Median $\overline{X}_g$ = a value (X) of the ordered DFs such that half of the data are greater than and half less than the value.

Table 1 lists the decontamination factors realized at the different facilities visited. The average decon DF is an arithmetic mean of the reported DFs. As can be seen the decon DFs range from 1.1 to 64 with the average decon DFs varying approximately by a factor of five. The average

Table 1

Decontamination Factors

Vendor	Plant	How Obtained	Range of Decon DFs	Average Decon DF $\bar{X}$**	Average man-rem DF $\bar{X}$**
1	A	Utility	3-31	10	3
2	B	Utility	2-30	10	5
3	C	Utility	1.1-64	23	5.4
2	D	Utility	N/A	N/A	3.5
2	E	EG&G TLDs EG&G instrument	8.1-14.9 6.4-25	10.3 11.8	4*
2	F	EG&G TLDs EG&G instrument	2.6-17 1.4-32	10.8 13.5	2.5*
2	G	Utility	1.8-17.5	4.1	N/A
5	H	Utility	N/A	N/A	1.5 & 6.5 (1)

* preliminary plant calculations
(1) Hot leg and cold leg DF's
N/A No data available.
** arithmetic mean.

man-rem DFs range from 1.5 to 6.5. It is interesting to note that even though plant B had a decon DF similar to plant A the man-rem DF more closely agrees with plant C. The reason(s) for this anomally can generally be characterized by these factors, (a) location of survey; (b) survey instrumentation and use; and (c) physical condition of the equipment surveyed. Tables 2 and 3 show examples of how an average DF can vary depending on which technique the vendor or licensee chooses. In Table 2 the average decon DF varies by a factor of 10 while the man-rem DF varies by 1.6. If the decon DFs are grouped by various regions a better determination of the effectiveness of the decontamination can be made and the variability between the average DFs, as calculated using the different methods, decreases. Table 3 lists similar information as Table 2 except for a different plant. The method of calculating the average DF for plant E has less variability due to a more uniform system decontamination.

Methods used in determining the decon and man-rem DFs have varied widely between facilities. Decon DFs have been determined using measurements taken with (a) unshielded portable survey instruments, (b) 2 pi shielded portable survey instruments, (c) portable survey instruments with collimated shielding, and (d) by EG&G using TLDs. Many facilities have used survey points that were established several years ago as part of a General Electric study on the rates of contamination or crud buildup inside of the recirculation piping. Man-rem DFs have been determined using (a) man-rem expended in performing similar pre and post-decon operations, (b) man-rem expended in performing a similar task from previous years and post-decon, and (c) direct measurements of general areas using portable survey instruments pre and post-decon. All of these different methods, particularly measurements made to determine a decon-DF, have several factors which can significantly affect the DF determinations. These factors can be catagorized as a) instrument selection, b) physical and spatial configurations of the equipment surveyed, c) survey point selection, d) consideration of background effects, and e) exactly matching stay times from previous years or pre and post decon operations.

Projections of man-rem savings are listed in Table 4. The man-rem savings have been significant with an average savings equal to 2112 with the range being 790 to 3660. Several plants have completed the operations for which the chemical decontaminations was performed as noted in Table 4, and in all cases the actual man-rem expended was less than the value initially projected with a decon. Plant experience has been that the scope of work was enlarged by addition of many non-scheduled items that would enhance future plant operation without causing the total man-rem to exceed the initial projected man-rem with a decon. Therefore, the actual man-rem savings is larger than initially projected. If the average cost of the decon vendor and waste handling is $1,000,000 the cost per man-rem saved is $474 using the 2112 man-rem figure. However, the dollars saved can be very significant and values ranging in the millions of dollars have been reported by various facilities.[4,5,6] In all cases the man-rem expended in performing the chemical decon and waste handling has been a small fraction of the total man-rem saved. The largest value of man-rem expended to date has been ~ 180 man-rem.

Examples of DF Variability   Table 2

Plant C

	Range	$\overline{X}$	$\overline{X}_g$	$\overline{X}_H$
Man-rem	1.9 to 18	5.4	3.3	3.3
Decon	1.1 to 64.3	23	11	2.4

$\overline{X}$, $\overline{X}_g$ & $\overline{X}_H$ by regions of the recirculation system

	Risers	Ring Header	Suction/Discharge	RHR	Bypass
$\overline{X}$	12	1.8	55	4.3	5.5
$\overline{X}_g$	7	1.8	58	4.3	5.5
$\overline{X}_H$	15	1.9	54	2.6	1.3

Examples of DF Variability   Table 3

Plant E

	Range	$\overline{X}$	$\overline{X}_g$	$\overline{X}_H$
Man-rem	1.5 to 5.3	3.4	3.7	3.2
Decon	6.4 to 25	12	10	10

$\overline{X}$, $\overline{X}_g$ & $\overline{X}_H$ by regions of the recirculation system

	Risers	Ring Header	Suction/Discharge	Lower Elbows
$\overline{X}$	9.2	25	12	9.4
$\overline{X}_g$	9.2	25	10	7.4
$\overline{X}_H$	9.2	25	12	8.1

ALARA Projections Table 4

Plant	Projected man-rem without decon	Projected man-rem with decon	Man-rem saved
A	3841	898	2943
A	3841	767[1]	2812
B	2654	426[2]	2228
C	2857	2000	827
D	*	*	1400[3]
E	935	145[2]	790
F	5436	2245	3190
G	1624	464[2]	1160
H	*	*	3660

1. This value includes drywell surface decontamination.

2. Actual man-rem expended for completion of job.

3. Initial man-rem projection, however actual job performed was greatly reduced in scope.

The normal waste forms generated have been cement solidifed resins. However, in two decons liquids were solidified with cement and in one operation the resin was solidified using the DOW system. Solidification vendor data[7] are listed in Table 5. The average volume of waste is 300 ft^3. However, plants A and C had situations arise which required much more waste to be generated. In one case concentrating equipment failed to function properly, thereby causing both liquid and resin wastes to be generated. And in the other case the decontamination solutions were not regenerated, causing more anion resin waste to be generated than would have been had the solutions been regenerated. Therefore, if these two plant values are excluded an average waste volume of 137 ft^3 results.

Table 6 is a preliminary list of general problem areas that have been identified by either plant or EG&G personnel. As can be seen most facilities have experienced minor problems with the vendor skids, and gasket, valve or seal failures. Most of these problems were easily corrected without significant schedule delays (i.e. less than twelve hours). The other problems areas that have been identified have caused greater delays. In some cases several factors have worked together to cause delays of several weeks. These longer term delays have cost the utilities valuable critical path time which can be related to millions of dollars. In general, the more significant problems have occurred with new vendors, particularly with the vendor decon skids. However, plant D shows that using experienced vendors is no guarantee that the decon effort will be free of problems. In nearly all cases the actual chemical decontamination (i.e. chemical injection to cleanup) effort progressed smoothly and required three to five days to be performed.

Tables 7 and 8 lists the results of some of the confirmatory and supplemental measurements that have been made by EG&G Idaho personnel. These measurements were made using the Eberline E530/HP22C A GM and different types of TLDs. The TLD measurements were made using "Hemi", personnel, and "Cheerio" TLDs. The "Hem*" TLD has been designed to measure beta exposure by incremental energy steps as well as total gamma exposure. The personnel TLD has been designed to measure only gamma exposure. The licensee measurements listed in Table 7 were made with a similar Eberline survey instrument, however their instrument was calibrated to ^{137}Cs while EG&G's instrument was calibrated to respond as though 80% of the source was ^{60}Co.

The results of the confirmatory measurements listed in Table 7, are in good agreement, and well within the error of the readings obtained using the Eberline GM survey instrument. Meter needle movement varies significantly with these instruments in low fields (i.e. < 500 mR/hr) because of the small GM detector probe used. Our preliminary results have not shown significant advantage to calibrating the survey instrument to respond as though 80% of the source was ^{60}Co.

The results of the supplemental measurements listed in Table 8 indicate that TLDs could be used to determine the pre and post decontamination exposure rates, and hence DFs, with little effect on the DF calculated at each location. The use of TLDs offers the advantage of less man-rem

Table 5

Solidified Waste Volumes

Facility	Decontamination Vendor	Waste Process(2)	Waste Volume(1) (ft^3)
A	1	CS	560
B	2	CS	160
C	3	CS	1020
D	2	CS	120
E	2	CS	160
F	2	CS	80
G	2	CS	120
H	5	DOW	180

1 Based on total liner volume
2 CS = Cement Solidification
DOW = DOW Chemical Solidification

Table 6

GENERAL TOPICS FOR PROBLEMS ASSOCIATED WITH DECON EFFORTS

Plant	Vendor	Hoses	Gaskets Seals or Valves	Minor Vendor Skid Failures	Major Vendor Skid Failures	Process or Chemistry Problems	Waste Handling	Scheduling	Personnel	
									Craft	Administrative
A	4	X	X	X	X		X	X	X	
B	2		X	X						
C	3	X	X	X	X	X	X			
D	2	X	X	X	X			X	X	X
E	2			X						
F	2		X							
G	2		X	X		X				
H	1				X	X		X		

Table 7 Results of Confirmatory Measurements

Licensee's Exposure Rate (Survey Instrument)	EG&G's Exposure Rate (Survey Instrument)	EG&G's Exposure Rate Average of TLD's
130	140	136
400	360	358
250	250	278
20	15	ND
40	65	ND
60	50	ND
160	ND	152
250	ND	240
250	ND	241

ND = No Data

Table 8 Results of Supplemental Measurements

EG&G's Exposure Rate (Survey Instrument)	EG&G's Exposure Rate (TLDs)	Ratio Survey Instrument / TLDs
146	134	1.09
148	152	0.97
245	240	1.02
248	241	1.03
225	274	0.82
290	486	0.60
840	706	1.19
540	497	1.09
425	358	1.19
58	63	0.92
65	67	0.97
65	61	1.07
39	35	1.11
40	40	1.0
775	782	0.99
775	912	0.85
615	595	1.03
388	370	1.05
430	458	0.94
1600	1582	1.01
178	171	1.04
	$\bar{X}$	1.00 ± 0.13

expended in making the pre and post decontamination measurements. EG&G's experience has been that it only takes one individual less than one hour in the drywell to place and remove the TLD whereas the survey instrument measurements require two men two to three hours to make the same measurements.

Results of the measurements made using the Ortec CPD-1 germanium detector are still preliminary. The detector is being calibrated for the numerous geometries that have been measured. These measurements have been made to assist in measuring DFs as well as to study the recontamination rate following a decontamination. Radionuclides observed in the gamma scans of the pipes have been ^{60}Co, ^{58}Co, ^{65}Zn, ^{54}Mn, and ^{137}Cs.

Samples of resins or liquids and their respective solidified forms have been obtained at plants A, E, and F as part of another NRC program. The solidified samples will be subjected to tests to determine the specific radionuclide leaching rates from the waste form.

Future plans for the program are to visit several more facilities to observe the decontamination processes with emphasis on facilities that are using different decon vendors or processes. Information obtained at each facility will be similar to past programs practices. Measurements will be made that will be used in evaluating the DFs as well as providing baseline data for determining recontamination rates. Final data and information will be gathered from the facilities where decons have already been observed. Also, more solidified waste samples will be obtained for the leachability study of these decon waste forms.

<u>Summary</u>

Determination of DF needs to be evaluated in greater detail. It should be determined if there can be a uniform method applied to obtaining the exposure dose rate values for pre and post-decontamination, as well as survey points and calculation methodology for obtaining an average DF value for the decon. This would assist in better understanding within the industry when results of chemical decontaminations are discussed.

Preliminary results have shown that in-situ chemical decontamination is an effective means of reducing total man-rem dose for recirculation pipe replacement, inservice inspections, and steam generator repair. Also the man-rem expended in performing the decontamination is a small fraction of the man-rem saved by performing the decon.

The normal waste form generated to date is cement solidified resins. Also, industry has indicated that the normal waste volumes generated from these chemical decontamination efforts are acceptable (i.e., the waste volumes are not excessive when compared to the normal volume of plant operational wastes generated).

Those normal problems that have been identified with chemical decontamination operations are not serious and have generally been solved on a plant by plant basis. However, significant problems have become apparent with some of the decon operations and the lessons learned from these problems need to be communicated within the industry. One lesson learned is, in EG&G's and industry personnel's opinion, that better quality assurance overview of the decon vendor skids, as they are initially assembled, would minimize problems at the licensee's site.

Preliminary data indicate that measurements made with similar survey instruments and TLDs are within the accuracy of the instruments. Also TLDs could be used to measure the pre and post decon exposure rates and save man-rem dose to licensee personnel. The gamma spectral data still require more work, however the radionuclides observed have been ^{60}Co, ^{58}Co, ^{65}Zn, ^{137}Cs, and ^{54}Mn. The gamma spectral measurements acquired to date will be used to evaluate radionuclide DFs as well as specific radionuclide behavior on a limited basis, as well in conjunction with the recontamination study.

The "Effectiveness and Safety Aspects of Selected Decontamination Methods" Program will visit more facilities this Fall and next Spring. More TLD and dose rate confirmatory measurements will be made. Measurements of specific radionuclide removal will be followed using a collimated intrinsic germanium detector system. Recontamination rates will also be studied using these varied measurement techniques. Criteria for determining DFs will be established. A data base that all utilities can use in estimating man-rem savings will be compiled. Future recontamination of the decontaminated systems will be studied. Lastly, in conjunction with another NRC study on solidifed waste leaching, samples of the waste and waste forms generated from these chemical decontaminations will be obtained.

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APPLICATION OF RELIABILITY METHODS IN ONTARIO HYDRO

R. Jeppesen

Ontario Hydro

1.0 INTRODUCTION

Ontario Hydro is a publicly owned Canadian utility which supplies electrical power to the Province of Ontario. It's installed capacity is currently about 22000 MW; peak power demand in 1983 was about 19000 MW (See Figure 1). It manages the design and construction of all additions to the bulk electricity system, including generating, transmission and distribution facilities.

Ontario Hydro have a large installed and committed nuclear program, with 10 commercial units in-service, totalling in excess of 6000 MW, with 10 further units to come in-service by the end of the decade, bringing the total to about 14000 MW (Figures 2 and 3). All reactor units are based on the CANDU, heavy water moderated and cooled, horizontal pressure-tube type reactor, incorporating on-power refuelling.

2.0 RELIABILITY PROGRAM MANAGEMENT

As Ontario Hydro's nuclear program developed (and indeed, in the development of other areas of plant design and operation) a "Quality Engineering" program has evolved based in large part on maintaining and improving plant and equipment reliability. This program stresses:

- clearly establishing design requirements including identifying quantitative performance targets for equipment;
- identification and correction of problems on a priority basis, to improve performance;
- feedback of operational experience to improve designs;
- planning and controlling work activities; and most importantly
- clearly identifying line management responsibility for achieving program goals.

The program contains many of the elements found in Quality Assurance programs and, in the nuclear area, it responds to Canadian jurisdictional QA requirements. The program is directed by a senior management committee, representing all line managers with responsibility for design, equipment procurement, construction, commissioning and operation of bulk electrical system facilities. It is chaired by the Vice-President, Design and Construction and reports to the Corporate Executive Office (Figure 4). This management aspect is considered to be a key element in promoting and maintaining an awareness of safety and reliability goals and ensuring that proper attention is given to providing a safe and reliable product to our customers.

Although the establishment of reliability programs which flow from this committee's direction is considered important to the broad reliability goals of Ontario Hydro, the aspect of instilling and maintaining a reliability "attitude", promoted directly by line managers, is perhaps the most important element in the program.

3.0 RELIABILITY PROGRAM

3.1 General

As mentioned, the program contains elements related to establishing quantitative reliability goals, measuring performance against those goals and providing feedback of equipment and system performance to improve both existing and new plant. The program has been developed and applied extensively in nuclear generating stations because of the recognized cost savings inherent in operating capital-intensive equipment at high capacity factors. The focus on nuclear plant safety also provides a major incentive to ensure reliability in safety-related equipment and systems. The following two sections indicate how reliability based performance targets - in both the safety and production areas - have been developed and how these are used in design and in operations to maintain and improve performance.

It should be stressed that a wide range of reliability techniques and methods can be applied. Our program stresses the selective application of these techniques to ensure we are paying proper attention to the right things. Although Ontario Hydro have a large integrated engineering capability, we believe it would be inappropriate to require a blanket approach to "reliability" for all plant systems and equipment or, indeed, even to all "safety-related" equipment. The reliability methods applied to plant systems and equipment are commensurate with the nature and complexity of the equipment. Sophisticated reliability methods applied to simple equipment and systems could well be counter-productive in terms of diluting limited resources.

3.2 Safety Reliability

3.2.1 Establishing Reliability Targets

The Canadian safety philosophy and the regulatory approach to safety requirements has considered risk-based criteria since its inception[1][2][3]. The first small demonstration plant built in Canada had quantitative reliability goals, established by the designer, for the most critical safety systems in the plant (Figure 5). These were derived based on the risk of a significant accident such that the risk was substantially less than that associated with the operation of a commercial coal-fired generating station. Our second nuclear generating unit (a 200 MWe prototype), also had such targets established, based on limiting the risk to the most exposed public individual (Figure 6). For the first commercial units and subsequent units, the Canadian Atomic Energy Control Board established a risk based "Siting Guide" from which targets for equipment reliability could be derived (Figure 7 and 8). (It also required that two classes of accident situations be analyzed and, in addition, specified individual and collective exposure limits for such situations.)

The guidelines and derived safety reliability targets for the first four Ontario Hydro nuclear stations are summarized in Figure 9.

In addition, analytical investigations by designers evolved from providing designs for selected systems which could meet the established reliability requirements, and assessing them against those targets, to the assessment, using detailed reliability techniques, of a range of more complex equipment failures. For Darlington GS, a comprehensive Probabilistic Safety Evaluation is being carried out as a design verification activity to assess plant wide response to a wide range of equipment failures (Figure 10). In addition to being a useful design verification, this assessment will also provide a valuable input into the production of operating procedures to deal with certain types of equipment failures and will provide an overall framework within which regulatory requirements can be addressed. Note that it is not intended or expected that further system or equipment reliability targets will be developed as a result of this exercise.

3.2.2 Monitoring In-Service Performance

For selected systems important to safety, a test program and data collection system is established in each operating plant. Actual component failure data is used to both report system reliability performance and to predict expected performance. These reliability assessments are also used to compare actual system performance with established system reliability targets. For systems with substantial redundancy such as shutdown (or scram) systems, simplified reliability models are used to assess the expected effect on system reliability of failures which (since they occur in redundant trains) do not contribute to actual system unavailability. This type of assessment and reporting is done for a limited number of systems, primarily those which are poised or on standby (e.g. Shutdown Systems, standby power, etc), for which (unsafe) failures are normally only detected during testing (See Figure 11). Information on these systems is collected and reported quarterly, with a more detailed evaluation done on an annual basis. The results are reported to the Atomic Energy Control Board but, more importantly, are used to establish programs for improvements, in current or future designs, if required.

3.3 Production Reliability

3.3.1 Reliability Targets

Overall plant and system targets are also established, early in the design, for all major plant process systems. These are based on past plant and equipment performance and on judgment of expected improvements. At this stage, a value for reliability (i.e., $ per percent change in capacity factor) is also derived for use by designers in life cycle costing analysis. This allows a rational trade off of capital costs vs life time operating costs. The targets are developed, again on a selective basis, into related equipment targets which may then be specified in technical specifications to manufacturers (see Figure 12). Direction is also provided to equipment suppliers to ensure they are placing the proper emphasis on the reliability of their equipment. This may consist of ensuring that they have taken into account significant operating experience on (their) similar equipment to, for complex equipment, requiring a reliability assessment (FMEA, Fault-Tree analysis, etc.) of their equipment. Again, it is stressed that a blanket approach to reliability requirements and assessments is not productive. Reliability analyses would not be appropriate or necessary for off-the-shelf components with a proven track record.

3.3.2 Monitoring In-Service Performance

During operation, broad reliability goals are established for plant performance, including capacity factor, outage rates, etc. (see Figure 13). Information is collected to track those systems and equipment which contribute to plant outages, as a guide to prioritizing maintenance work and to identify needs for design or operating improvements (see Figure 14). This information is also used to establish reliability targets for new designs (see Figure 15 and 16).

3.4 Information Feedback

One significant activity related to any reliability improvement program is the collection, distribution and use of information gathered during the construction, commissioning and operating phases of plants.

Ontario Hydro have stressed from the outset, not only collecting and assessing operating information and providing it to all stakeholders, but in general, promoting close cooperation and communication between designers, operators and equipment suppliers. This was of particular importance, since while our nuclear program has evolved to larger units, it has been based on a single concept - the CANDU PHW reactor. Experience gained on our demonstration and prototype plants has been directly applied by design and operating groups to effect significant improvements in present commercial units.

In addition to promoting and supporting the use of operating experience with all suppliers, we have taken major initiatives to feedback information directly to suppliers of major equipment (steam generators, turbine-generators, large pumps).

Ontario Hydro's approach to design and operation, including the use of reliability techniques has produced excellent results (see Figure 17). Our two major four unit stations, Pickering A and Bruce A have lifetime capacity factors of about 80% and 85% respectively (our best unit - Bruce 3 (750 MWe) - has a lifetime (6 years) capability factor of about

90%). Although recent difficulties with pressure tubes on two of the four units at Pickering A have substantially reduced performance at that station, the four units at Bruce A, to the end of August have averaged 96% capacity factor, and the first unit of Pickering B, declared in service in May, 1983 has achieved a capacity factor of 80% from that date to September 30.

4.0 CONCLUSION

Reliability techniques can provide a useful management tool for achieving and maintaining high levels of safety and plant production. The techniques are used primarily to focus on significant contributors to equipment unavailability so that appropriate attention is paid to correcting any deficiencies.

The methods and techniques used can range from simple evaluation of past performance to detailed reliability modelling. Ontario Hydro emphasizes line management's responsibility to achieve established reliability goals and to exercise the necessary judgment in applying appropriate resources and methods to achieve those goals.

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FIGURE 1

ONTARIO HYDRO

GENERATING CAPACITY - 1983

TYPE	INSTALLED CAPACITY MW	%	ENERGY GENERATED %
HYDRAULIC	6,500	30.5	33
FOSSIL	9,200	43.2	32
NUCLEAR	5,600	26.3	35
TOTAL	21,300	100	100

PEAK POWER DEMAND IN 1983 18,792 MW

FIGURE 2

NUCLEAR CAPACITY

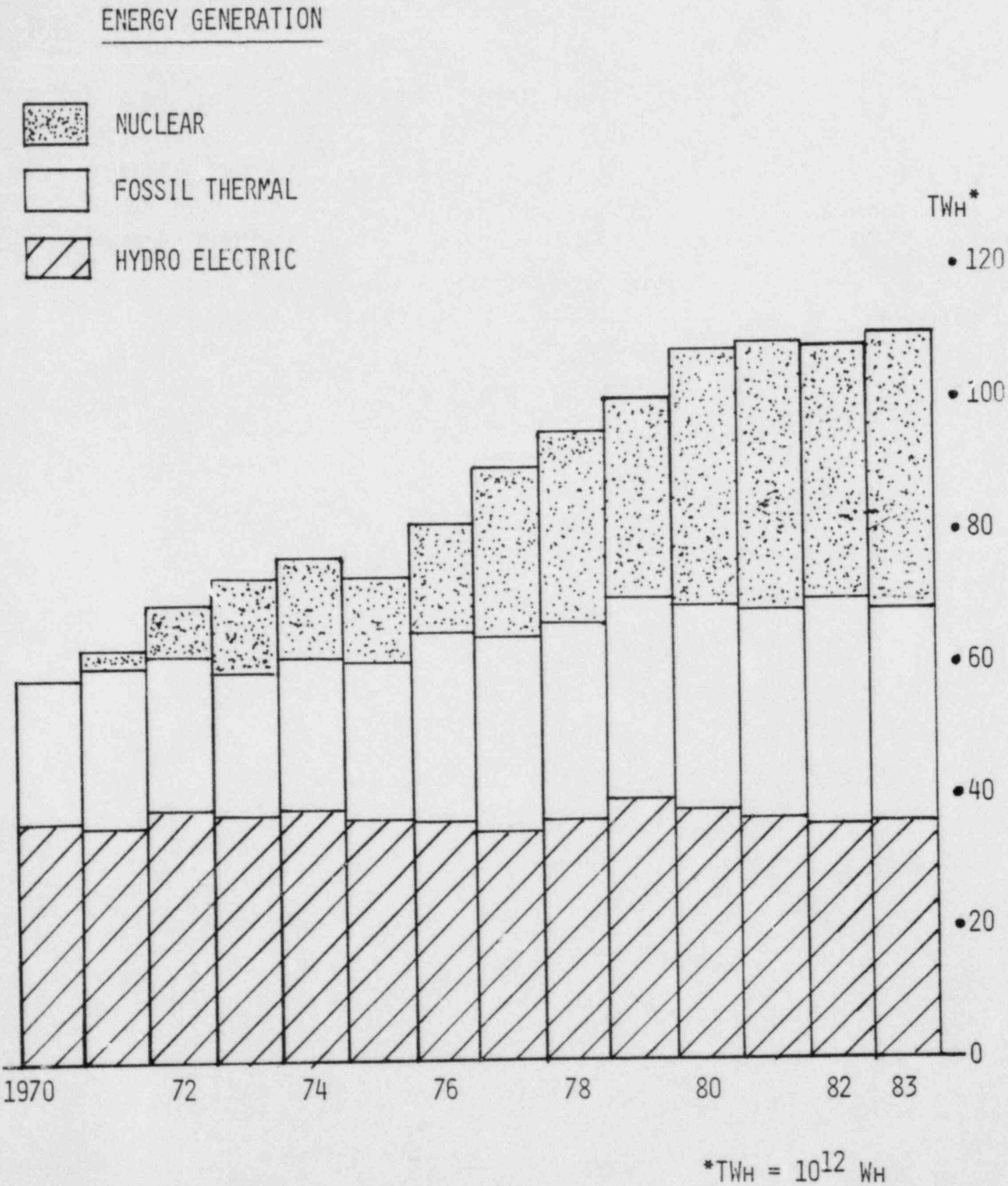


FIGURE 3

ONTARIO HYDRO NUCLEAR GENERATING STATIONS

NAME	CAPACITY (MW)	IN SERVICE DATE
NPD	25	1962
DOUGLAS POINT	218	1968
PICKERING A	4 x 540	1971 - 1973
PICKERING B	4 x 540	1983 - 1985
BRUCE A	4 x 826	1977 - 1979
BRUCE B	4 x 826	1984 - 1987
DARLINGTON A	4 x 850	1988 - 1990

FIGURE 4

Ontario Hydro Quality Engineering Organization

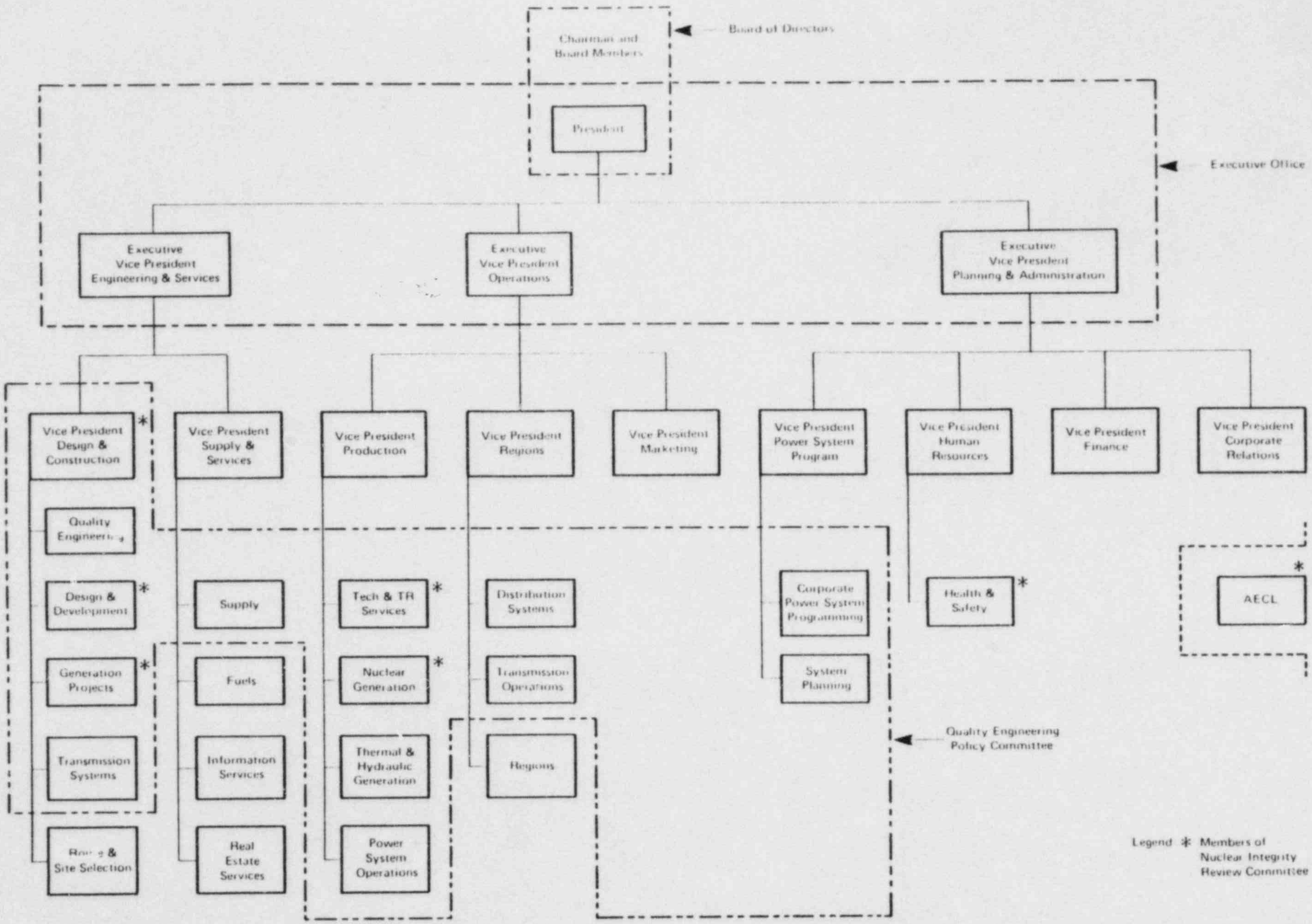


FIGURE 5

NUCLEAR POWER DEMONSTRATION REACTOR

(25 MWe - 1962)

CRITICA - 10^{-2} FATALITIES/YEAR

↓

RISK OF MAJOR ACCIDENT - 10^{-5}/YEAR

↓

RELIABILITY TARGETS

E.G. CONTROL SYSTEM - 10^{-2}/YEAR

SCRAM SYSTEM - 10^{-4} YEAR/YEAR

ECC SYSTEM - 10^{-2} YEAR/YEAR

FIGURE 6

DOUGLAS POINT (218 MWe - 1968)

CRITERIA - RISK TO INDIVIDUAL
10^{-6}/YEAR

↓

RELIABILITY TARGETS

E.G.	SCRAM SYSTEM	-	10^{-3} YEAR/YEAR
	CONTAINMENT PROVISIONS	-	10^{-2} YEAR/YEAR
	ECC SYSTEM	-	10^{-2} YEAR/YEAR

FIGURE 7

PICKERING (4 x 540 MWe - 1971)

SITING GUIDE

CLASS OF FAILURE	FREQUENCY	CONSEQUENCE INDIVIDUAL	POPULATION
SINGLE FAILURE	1/3 YEAR	0.5R - W.B. -	10^4R
		3R THYROID -	10^4R
DUAL FAILURE	1/1000 YEAR	25R - W.B. -	10^6R
		250R - THYROID -	10^6R

FIGURE 8

BRUCE (4 x 826 MWe - 1977)

SITING GUIDE

MODIFIED FROM PICKERING

- FREQUENCY OF 'DUAL FAILURE'

  1/3000 YEAR

- 2ND INDEPENDENT SHUTDOWN SYSTEM ADDED

FIGURE 9

SUMMARY OF RISK BASED RELIABILITY TARGETS

	BASIC RISK CRITERIA	DERIVED RELIABILITY TARGETS (TYPICAL)	
NPD	10^{-2} FATALITIES/YEAR	SHUTDOWN SYSTEM	10^{-4}(UNAVAILABILITY)
		CONTROL SYSTEM	10^{-2}/YEAR
		ECI SYSTEM	10^{-2}(UNAVAILABILITY)
DOUGLAS POINT	INDIVIDUAL RISK $<10^{-6}$/YEAR	SHUTDOWN SYSTEM	10^{-4}(UNAVAILABILITY)
		ECI SYSTEM	10^{-2}(UNAVAILABILITY)
		CONTAINMENT	10^{-2}(UNAVAILABILITY)
PICKERING	SITING GUIDE	SHUTDOWN SYSTEM ECI SYSTEM CONTAINMENT	3×10^{-3} (UNAVAILABILITY)
BRUCE	REVISED SITING GUIDE	SHUTDOWN SYSTEM ECI SYSTEM CONTAINMENT	10^{-3}(UNAVAILABILITY)

FIGURE 10

CURRENT APPROACH

DARLINGTON PROBABILISTIC SAFETY EVALUATION (DPSE)

- OVERALL PLANT SAFETY ASSESSMENT
- SEPARATE PROJECT TEAM, INCLUDING DESIGNERS FROM EACH DISCIPLINE AREA

RESULTS

1. HAS IDENTIFIED SEVERAL DESIGN SHORTCOMINGS - GENERALLY EASY TO FIX.
2. IS BEING USED TO GENERATE DETAILED OPERATING PROCEDURES FOR ABNORMAL PLANT TRANSIENTS.

FIGURE 11

BRUCE NGS-A
SAFETY SYSTEM PERFORMANCE TRENDS UNAVAILABILITY

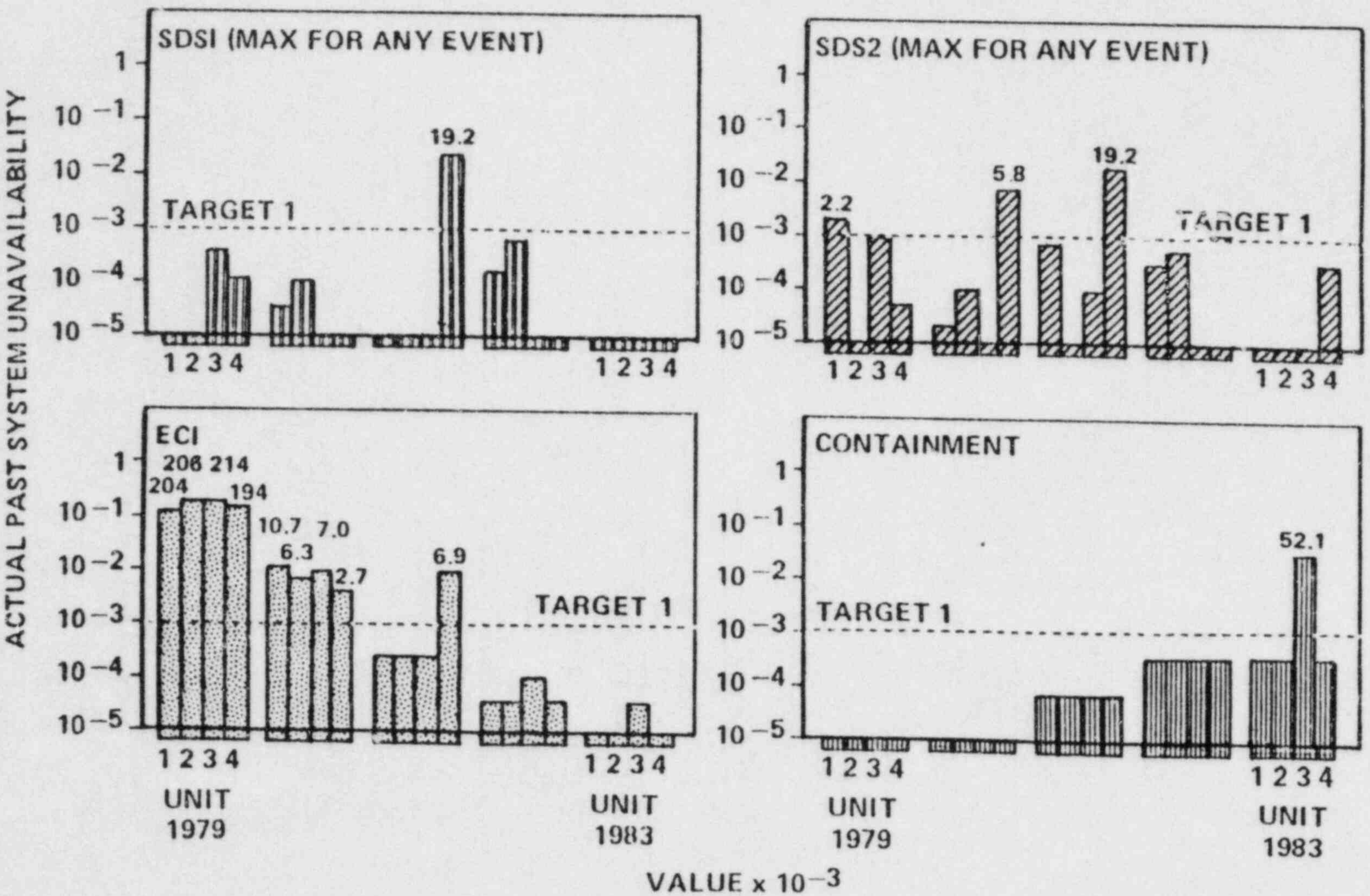


FIGURE 12

PRODUCTION RELIABILITY ANALYSIS APPROACH

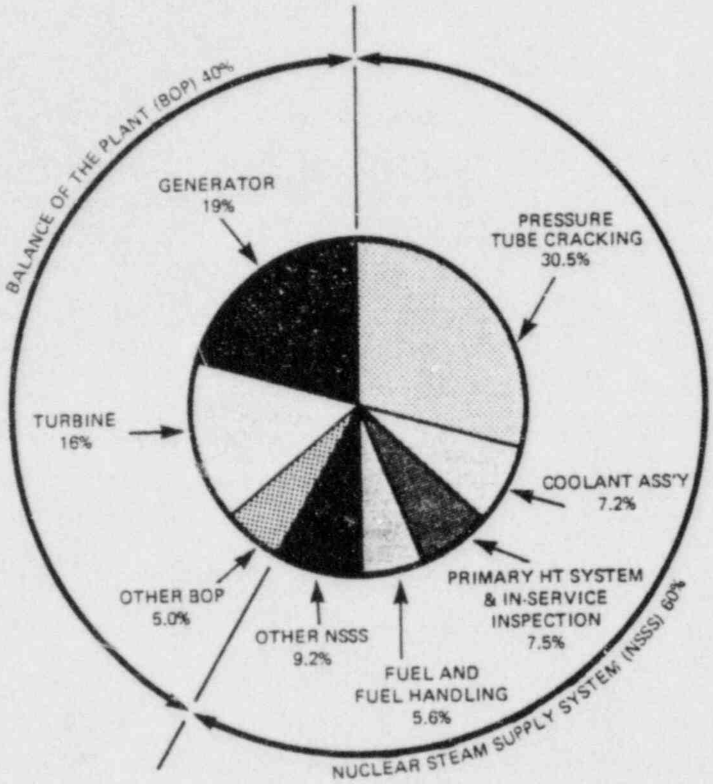
	EXAMPLE (DARLINGTON GS 850 MW)
ESTABLISH UNIT TARGETS	• UNIT EFOR = 6.0% • UNIT ICbF = 14.5%
↓ ESTABLISH MAJOR SYSTEM TARGETS	TURBINE-GENERATOR & AUXILIARIES EFOR = 1.3% ICbF = 1.7%
↓ ESTABLISH MAJOR EQUIPMENT TARGETS	TURBINE-GENERATOR SET EFOR = 1.1% ICbF = 1.6%
↓ SPECIFY TARGETS (TECHNICAL SPECIFICATIONS)	TECHNICAL SPECIFICATIONS FOR TURBINE-GENERATOR SET
↓ SPECIFY EQUIPMENT R&M PROGRAM REQUIREMENTS IN TECH. SPEC.	• ANALYSE OPERATING INFORMATION • IMPROVEMENT PROGRAMS • FAULT TREE ANALYSIS
↓ MANUFACTURE TO SPECIFICATION	• TEST • INSTALL • COMMISSION
↓ IN-SERVICE MONITORING	• TO IDENTIFY SPECIFIC MAJOR CONTRIBUTIONS TO DOWNTIME • TO COMPARE EXPERIENCE TO PREDICTED

FIGURE 13

OPERATING TARGETS - BRUCE A (4 x 826 MWe)

PARAMETER	TARGET - 1984	EXPERIENCE TO END OF AUGUST 1984				
		UNIT 1	2	3	4	STATION
NET CAPABILITY	88%	92.8	95.9	98.5	98.2	96.3
NET CAPACITY	88%	92.6	95.6	98.5	98.2	96.2
NUMBER OF SUDDEN FORCED OUTAGES	16	2	4	1	2	9

FIGURE 14



PICKERING GS 'A'
CAUSES OF INCAPABILITY (EXCLUDING 1972 STRIKE)
IN SERVICE TO SEPTEMBER 1978

FIGURE 15

850 MW NUCLEAR UNIT – AVAILABILITY TARGETS

type of outage	rate %	hours/yr.	no. of occurrences/yr. total	sudden
forced	4.5	363	8	2
equivalent forced	6.0	484		
maintenance	2.0	175	4	—
planned	6.0	526	1	—
total unavailability	12.1	1063	13	2

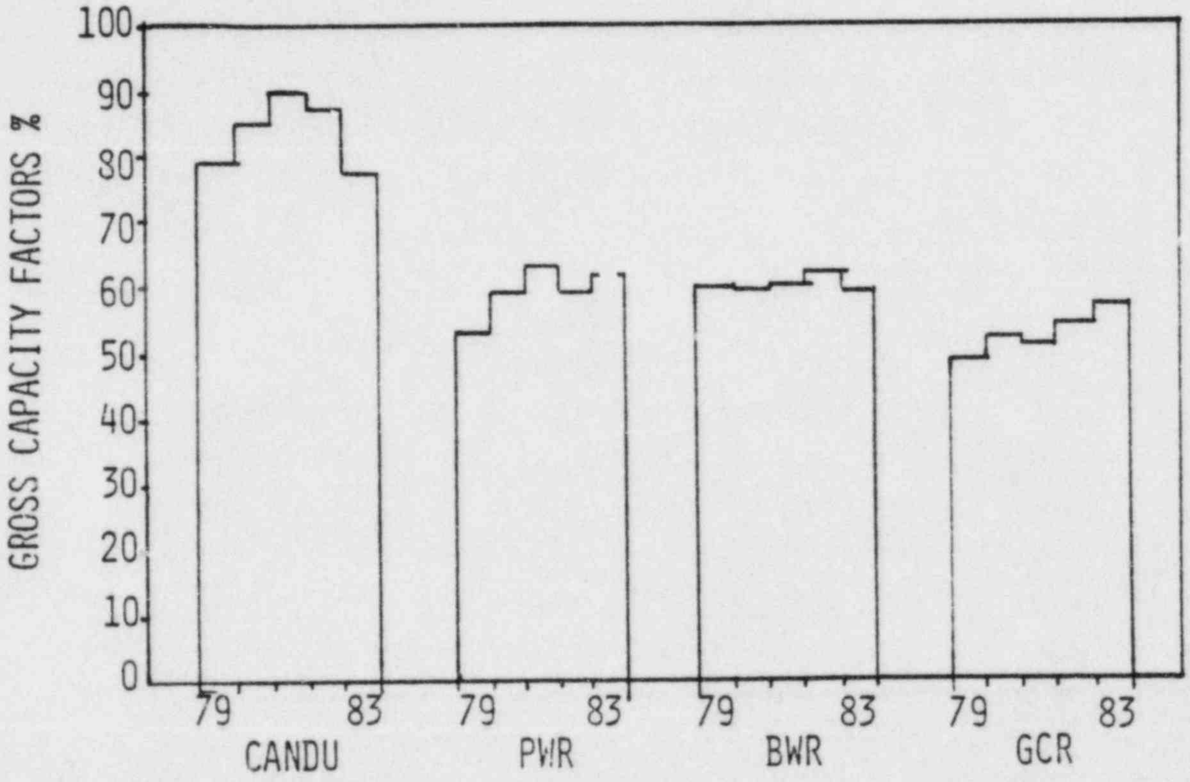
FIGURE 16

850 MW NUCLEAR UNIT – TARGETS FOR FORCED OUTAGE DURATION, RATE, AND FREQUENCY

System	Forced Outage Duration Hours/year	Rate %	Frequency No./year
General causes	3.2	0.04	0.06
Buildings and Structures	7.3	0.09	0.09
Reactor – Boiler and auxiliaries	189.4	2.35	2.43
Turbine – Generator & Auxiliaries	111.2	1.38	1.56
ELECTRIC POWER SYSTEMS	15.3	0.19	0.30
Instrumentation and control	28.2	0.35	3.32
Auxiliary Processes & Services (water & comp. air)	8.1	0.10	0.21
unit total	363.0	4.50	8.00

FIGURE 17

COMPARISON OF CAPACITY FACTORS WITH OTHER COMMERCIAL REACTOR TYPES LARGER THAN 500 MWe



AIRLINE EXPERIENCE WITH RELIABILITY-CENTERED MAINTENANCE

THOMAS D. MATTESON

AMERICAN MANAGEMENT SYSTEMS, INC.

A PRELUDE (1946-66)

The state of the art of air transport maintenance program designed immediately after WWII can best be described as a process based on an aggregation of practical experience, common sense and industrial folklore focused on ensuring the highest level of operating safety. It was a nascent state, not yet coherent, unstable, but trying to respond effectively to a rapidly growing and exciting industry.

It was a time for developing ideas:

- On-condition maintenance. Substitution of tasks that measured physical condition and compared it with a standard to determine the need for replacement or adjustment rather than time-directed overhaul or discard.

- Overhaul time extensions. Increasing the time between scheduled overhauls of engines and system components based on sample disassembly inspections of time-expired hardware.

- Regulation by statistics. (The first FAA try didn't work). Using simple statistical analyses of engine shutdowns or system component failures to control overhaul periodicity. (The small numbers of events resulted in sampling variations that obscured the real characteristics of the population, and the assumption that these failures were age-related was invalid).

- Reliability analysis. First, recognizing that the underlying premise of preventive maintenance was that there was an adverse age-reliability relationship; then designing a method for finding out whether this premise was true. The results of these analyses showed that for most complex assemblies it was not. We had thought that such analyses would provide a rational basis for selecting overhaul and discard periodicities (increasing the effectiveness of these tasks). Instead, they told us that most scheduled overhaul tasks were not applicable.

- Reliability programs. The airline's growing knowledge about maintenance program management led to agreement by the FAA to underwrite several in-service test programs. United Airlines chose the most imaginative tack. It established a process for tracking the reliability and physical condition of a group of complex mechanical components to determine their reaction to increases in time since overhaul. The lessons learned provided the basis for the first large scale age exploration program for complex assemblies in a new type of aircraft, the Boeing 737. This program replaced the traditional, intuitively selected overhaul intervals for more than 40 system components with a process for tracking in-service reliability. A scheduled overhaul task was required only if at some age there was a significant deterioration in reliability.

  Some airline reliability programs focused on reliability improvement, rather than maintenance program management by using pilot-report statistics to focus on in-service reliability problems.

The success of these programs opened the door for another giant step in the process for preventive maintenance program design.

But before describing that work, let's review the concurrent work in the area of structures.

- Structural sampling. Although we had conclusively demonstrated that complex devices rarely had adverse age-reliability characteristics, structures were recognized to be entirely different. The FAA design requirements for primary structure in transport aircraft and the fact that each large airline had a considerable number of nominally identical aircraft suggested that a sampling process might be a highly effective way of managing aging aircraft structures. Some early approaches suggested that a fixed percentage of the critical structural details be periodically inspected, but the method for selecting the samples was not described. The airline structures specialist had several problems. Ideally, he knew that the probability of finding structural defects increased with age; he also knew that early production aircraft had a higher risk of assembly preloads than later aircraft; he also knew that the designers' estimates of expected life (at that time) were not reliable. The resulting program design provided for periodic sampling inspections of each structural detail in which the sampling schedule ensured that before the expected life of each detail was reached, that all of those details had been inspected. The process, called constant density sampling, is illustrated in Figure 1.

The 1950s and 60s provided the time for acquiring a great deal of technical knowledge about maintenance program design. An important change in the administrative process also occurred.

- Maintenance Review Boards. Originally, the maintenance program for each new aircraft resulted from negotiation between the airline and its assigned FAA Maintenance Inspector. The advent of the jet airplane and its new technology resulted in a change in this process. The initial purchasers joined together, combining their experience and the knowledge of the manufacturer about his new design, to propose a common initial preventive maintenance program for all users and submit

CONSTANT DENSITY SAMPLING

AIRCRAFT ENTRY NO.	OVERHAUL VISIT 1	2	3	4
1	● ⚠ ①	● ◬ ②	● ⚠ ③	● ◬ ④
2	● ◬ ②	● ⚠ ③	● ◬ ④	● ⚠ ①
3	● ⚠ ③	● ◬ ④	● ⚠ ①	● ◬ ②
4	● ◬ ④	● ⚠ ①	● ◬ ②	● ⚠ ③
5	● ⚠ ①	● ◬ ②	● ⚠ ③	● ◬ ④
6	● ◬ ②	● ⚠ ③	● ◬ ④	● ⚠ ①
7	● ⚠ ③	● ◬ ④	● ⚠ ①	● ◬ ②
8	● ◬ ④	● ⚠ ①	● ◬ ②	● ⚠ ③
9	● ⚠ ①	● ◬ ②	● ⚠ ③	● ◬ ④
10	● ◬ ②	● ⚠ ③	● ◬ ④	● ⚠ ①

● - SSI, Group I - 100% inspections
△ - SSI, Group II - 50% inspections (subgroups ⚠ ◬)
○ - SSI, Group III - 25% inspections (subgroups ①②③④)

FIGURE 1

it jointly to the FAA for its approval. The FAA convened a Maintenance Review Board to review that proposal and negotiate an approved initial program. (After initiating operations, the individual operator/FAA relationship was re-established).

This process was highly effective in bringing more knowledge to the table, but it had a significant weakness - there was no rigorous process underlying the concensus of the operators, so that the conservative opinions of the FAA, although often less knowledgeable than the users', would prevail.

The apex of this problem appeared when, in a preliminary meeting to discuss the development of the maintenance program for the Boeing 747, the FAA advised that the size of that aircraft would, in their view, require a far more extensive and intensive program than any previous model. It was clear that this was an emotionally conceived position and that the airlines had a great deal of homework to do before proceeding.

A NEW APPROACH TO PREVENTIVE MAINTENANCE PROGRAM DESIGN (1967-68)

Having carefully acquired abundant information about the age reliability characteristics of complex assemblies, Stan Nowlan at United Airlines suggested the use of a decision tree to identify requirements for preventive maintenance. The first technical paper focused on hardware assemblies and asked a series of questions about the impact of characteristic failure.[1] This approach recognized that two kinds of functional failures were of interest, those affecting safety and those affecting economics. It further separated each of those into two sub-sets, those safety-related failures that were immediate and those that were hidden from the user; and those economic-related failures that were preceded by measurable, physical, or performance degradation and those that were not.

The decision-tree approach was accepted by the 747 Maintenance Steering Group and was the basis for a handbook called MSG-1 that was used to guide the development of the initial preventive maintenance program for the Boeing 747.

The FAA reaction to the size of the 747 and its potential influence on its maintenance program was the catalyst that created what we now call "Reliability-Centered Maintenance".

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[1]Matteson, T.D. and F.S. Nowlan, Current Trends in Airline Maintenance Programs, AIAA Commercial Aircraft Design and Operation Meeting, Los Angeles, CA, 1967 (AIAA 67-379)

WHAT IS RELIABILITY-CENTERED MAINTENANCE (RCM)?

RCM is a rational, coherent approach to the problem of preventive maintenance program design based on a decision tree. It has broad potential applications, because it focuses on functions and functional failures and their threats to operations, not on peculiar qualities of the design or environment.

The RCM process requires:

- Identification of the systems, subsystems, and equipments for which a preventive maintenance program is desired.

- Exhaustive identification of the functions provided. The most frequently used level for this analysis is the subsystem - the first level of aggregation above equipment. Analysis at the system level is also required for functions that are the result of subsystem interactions. Analysis at the equipment level can be done selectively if desired, but experienced analysts usually find that subsystem-level analyses can address most functions.

- Exhaustive identification of functional failures - the ways in which loss of function can occur.

- Identification of the dominant causes of functional failures - either the result of failure absolutely requires that failures be prevented, or there are so many failures that it is economically worthwhile to prevent them.

- Identification of the impacts of these failures.

- Classification of each functional failure/dominant failure mode using a decision tree to determine if:

    - Evident - Safety Related
    - Evident - Operations Related
    - Evident - Support Related
    - Hidden

- Identification of applicable and effective preventive tasks.

  - Applicable - task works
  - Effective - task worth doing

- Identification of design changes when there is no applicable and effective preventive task for a safety-related failure.

WHAT KNOWLEDGE UNDERLIES RCM's DESIGN AND APPLICATION IN THE AIRLINE ENVIRONMENT?

RCM was designed in the airline environment because that environment demanded a high level of safety, and some airline managements were willing to invest in applied research to answer, for maintenance, that question posed by Pontius Pilate nearly 2,000 years ago, "What is truth?" One airline senior vice president asked, "Why do we do maintenance?" That airline applied inductive reasoning (i.e., the ability to reason from the particular to the general) based on ideas and experience from such widely diverse disciplines as design, operations, maintenance, rule-making and regulation, information systems, statistics and operations research to create what we now call RCM, an alternative to intuition and industrial folklore as a resource for designing preventive maintenance programs.

The knowledge underlying the development of RCM included:

- The characteristics of air transport design, including knowledge about the historical and contemporary regulatory requirements for the design of air transport systems and structures. The most important of these were the requirements for redundancy in both systems and structures.

- The characteristics of air transport operations, particularly the importance of maintaining schedule and the irretrievability of revenues lost as the result of delays or cancellations.

- The role of preventive maintenance as an ensuror of function only when there is an adverse age-reliability relationship that can either be measured in-service at the item level or verified by appropriate statistical analyses.

- The kinds of preventive tasks - hard time, on-condition, failure-finding, each having a specific role defined by the decision tree classification process.

- The qualities required of preventive tasks - applicability and effectiveness.

- The definition of "function" - the purpose of the design, either active or passive, and evident or hidden to the operator.

- The definition of "failure" - loss of function, which varies with the interest and objective of the observer.

- The effect of criticality - for evident failures, safety comes first, then the ability to operate (produce the product), then support; for hidden functions consideration of their criticality and their reliability and recognition that excessive testing decreases reliability.

- The value of system-level solutions - equipment-directed decisions tend to be sub-optimal. ("Pareto is not dead"; 80% percent of our problems come from 20% of the causes); the importance of the users view of safety and operations, not each equipment manufacturer's perception.

- The results of age-reliability analysis of a large number of quite different complex equipments as shown in Figure 2.

EXPERIENCED AGE-RELIABILITY RELATIONSHIPS

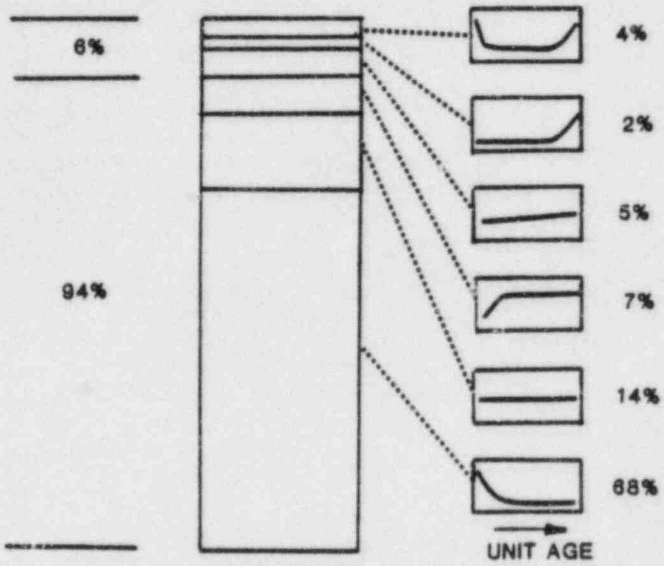

FIGURE 2

UNDERSTANDING AND APPLYING RCM

We must learn to think clearly about function. (Brakes don't stop cars -- they stop wheels. Braking systems stop cars. (The most difficult part of RCM is identifying functions). If you understand functions, then you can understand functional failure, the loss of function or degradation below operationally acceptable levels.

Failure models are quite different for simple items and complex items. Failures of single parts of mechanisms are usually age-related. Failures occur when their resistance to failure degrades to the level of some instantaneous stress. For a single iten of this kind, this relationship is shown in Figure 3. For several items of the same nominal kind, it is shown in Figure 4. For assemblies, unless the designer achieves the quality of design achieved in Oliver Wendell Holmes' wonderful one-horse shay, it is likely that failures will occur over a wide range of ages because of the differences in failure resistance of the component parts as shown in Figure 5.

These models help us to understand why we should not expect sharply define "wear-out" age-reliability characteristics in complex hardware, as shown previously.

The selection of preventive tasks requires that the analyst not only understand what can be done in that environment perceived (or at least yearned for) by the designer, but what will work in the real world and also be worthwhile. My work has uncovered so much that is inapplicable or ineffective but named "preventive", that I encourage you to look for yourself.

I found:

- A requirement to lubricate a standard 16MM sound motion picture project with 5 kinds of lubricant.

- A requirement to use an industrial pyrometer to routinely measure water temperatures in a dishwasher.

MODEL OF FAILURE PROCESS

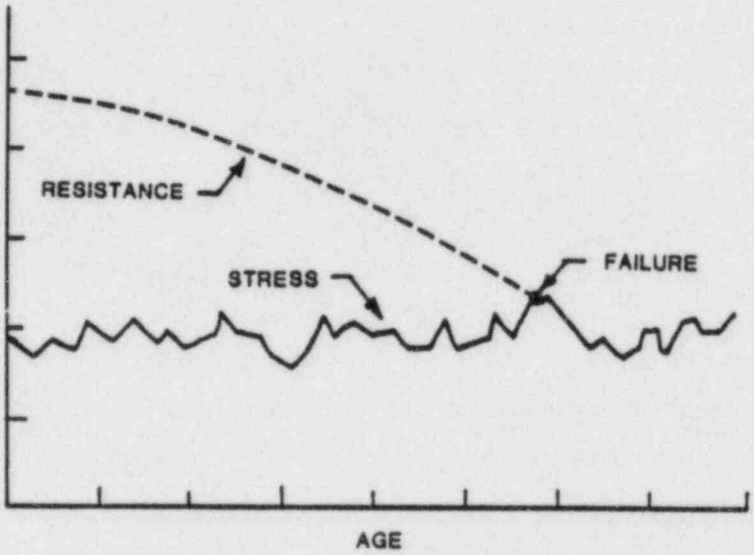

FIGURE 3

VARIATION OF AGES AT FAILURE–
SAME FAILURE MODE

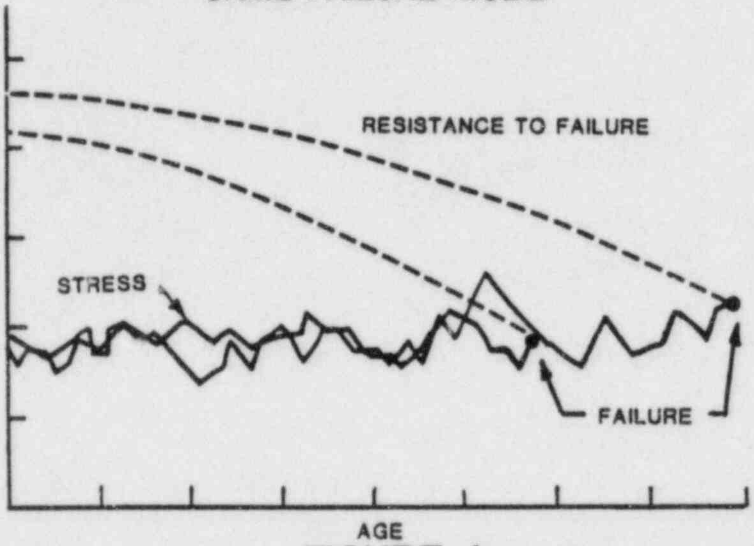

FIGURE 4

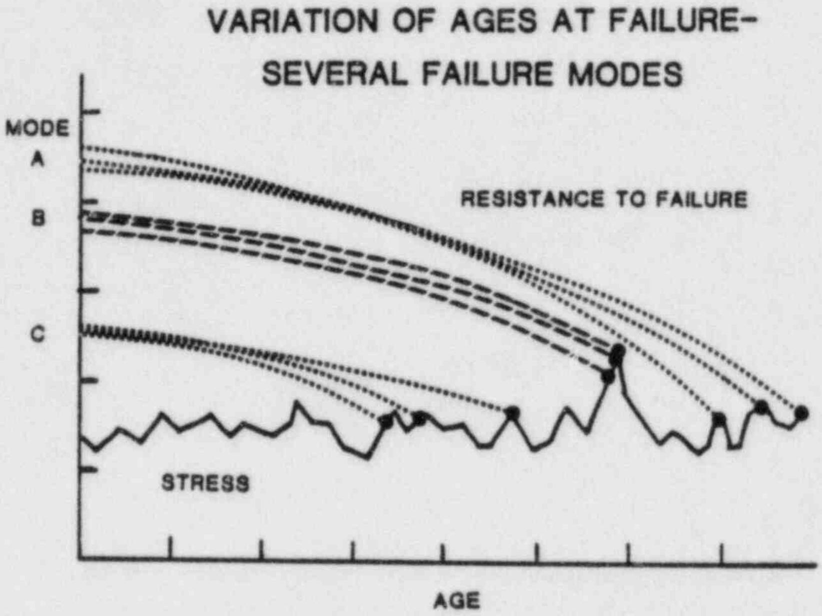
VARIATION OF AGES AT FAILURE-
SEVERAL FAILURE MODES
MODE
A
B
C
RESISTANCE TO FAILURE
STRESS
AGE

FIGURE 5

- A requirement to disassemble pump/motor drive couplings to access their condition instead of making a backlash check.

- A requirement to disassemble a sealed position indicator to see if the internal gears showed evidence of wear when the indicator's accuracy could be externally checked.

- Many task periodicities based on mean time between failures.

Preventive tasks must produce the desired effect in the environment in which they will be used. We call this quality "applicability". It is the primary requirement for every PM task. This requirement prevents the imposition of tasks that appeal to the analyst engineer but, on balance, in the real environment have either no impact or a degrading impact on the hardware.

Preventive tasks must be worth doing. We call this quality "effectiveness". If you ask a mechanic to perform a task more often than he perceives is necessary or require a complex, overly precise measurement technique you are wasting the resources of the enterprise.

Applying RCM requires a rigorous process:

- Partition and identification.

- Function and functional failure analysis.

- Failure modes and effects analysis (but not that required by MIL-STD-1629).

- Failure classification and task selection (the decision tree process).

- Safety-related design change recommendations.

The most difficult parts are the first two steps. "Familiarity breeds contempt." In this case, familiarity causes resistance to careful thinking, because the analyst believes he "already knows" the answers.

What results can you expect? If your technology has been led into preventive maintenance by poorly conceived regulations or by what I have loosely described as "industrial folklore", you can expect a significant reduction in PM tasks and some concurrent improvement in reliability. (As an example, the DC-8 had about 200 hard time overhaul tasks in its initial program, the 747 had less than 10.)

If your PM program has been built on a basis of hard-nosed shop experience with specific failures driving specific actions, you can expect some deletions and some new tasks, mostly failure-finding tasks for hidden functions not previously identified.

In addition, your analyses will identify opportunities for age exploration that will improve the effectiveness of important existing tasks and provide a rational basis for functional and regulatory support of changes to your preventive maintenance program.

APPLICATION OF NASA KENNEDY SPACE CENTER SYSTEM ASSURANCE ANALYSIS METHODOLOGY TO NUCLEAR POWER PLANT SYSTEMS DESIGNS

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INTRODUCTION

In May of 1982, the Kennedy Space Center (KSC) entered into an agreement with the Nuclear Regulatory Commission (NRC) to conduct a study to demonstrate the feasibility and practicality of applying the KSC System Assurance Analysis (SAA) methodology to nuclear power plant systems designs. North Carolina's Duke Power Company expressed an interest in the study and proposed the nuclear power facility at CATAWBA for the basis of the study. In joint meetings of KSC and Duke Power personnel, an agreement was made to select two CATAWBA systems, the Containment Spray System and the Residual Heat Removal System, for the analyses. Duke Power provided KSC with a full set of Final Safety Analysis Reports (FSAR) as well as schematics for the two systems. During Phase I of the study the reliability analyses of the SAA were performed. During Phase II the hazard analyses were performed. The final product of Phase II is a handbook for implementing the SAA methodology into nuclear power plant systems designs.

The purpose of this paper is to describe the SAA methodology as it appiles to nuclear power plant systems designs and to discuss the feasibility of its application.

BACKGROUND

The SAA methodology was developed by the Kennedy Space Center (KSC) to meet specific safety and reliability goals that are established for the Space Program. These goals required a plan for the systematic identification, tracking, resolution, and control of critical items (reliability problems) and system hazards (safety problems), both design and operational. The resulting plan describes a method for meeting these goals through the use of three key elements; a systematic analysis, a information system, and a review system. The plan required the integrating of reliability and safety analytical tasks to ensure that the groundrules developed for performing these tasks are complementary, preventing inadvertent omission of the identification of some critical items and hazards. A typical example of this form of omission is the FMEA groundrule that passive system elements are not considered in the reliability analysis. The integrated analysis method ensures that passive elements are considered in the safety analysis. Some other examples of groundrule omissions are structural failures, multiple failures, and human errors. In summary, the integration of the reliability analysis and the safety analysis is a response to the requirement to identify all critical items and system hazards.

The SAA methodology was developed for application to complex facilities that involve the operation of many unique systems. Nuclear power plant systems and aerospace launch support systems are similar in complexity of design and share common safety and reliability goals. This similarity prompted a study which demonstrated the feasibility and practicality of applying the SAA methodology to nuclear power plant systems.

The SAA methodology can be applied to the completed design; however, to reduce backfitting of hardware and ratchetting of regulations, it should be incorporated as part of the design engineering process. An SAA is performed on all systems in the facility. The results of the SAA for each system are tabulated in a summary document. A comprehensive cataloging of the results of the SAA's performed on each system in the facility provides clear visibility of reliability and safety concerns for review and action. The cataloging of itical items and system hazards and their associated resolutions provides e following:

a. Data indicating re-design is necessary to meet reliability goals (eliminate critical items).

b. Data indicating re-design is necessary to meet safety goals (eliminate hazards).

c. Data necessary to evaluate the resolution of residual critical items and assess the risk.

d. Data necessary to evaluate the resolution of system hazards and assess the risk.

SAA METHODOLOGY

The SAA methodology is simply the combining of well known safety and reliability analytical techniques with unique management elements to identify, track, and resolve critical items and system hazards. The key elements of the methodology are the analysis, the management system , and the review cycles.

DESCRIPTION OF KEY ELEMENTS.

Figure 1 identifies the objectives and key elements that comprise the total SAA methodology. The key elements provide for application of a disciplined analytical method, assessment and application of the results, and insured visibility of the results for consideration of any associated risk.

The System Assurance Analysis (SAA). The goal of a reliability analysis is the prevention of "loss of system" and "system function degradation". The goal of a safety analysis is the prevention of "loss of life" and "system safety degradation". The SAA is a technique for integrating the reliability and safety analyses to meet these goals. Figure 2 illustrates the simplicity of this technique to accomplish the SAA.

The SAA examines systems, subsystems, components, control functions, integrated systems, and human/machine operations. The reliability analytical techniques of the SAA assess and categorize system hardware elements by their failure effect on plant operation. These techniques result in the identification of hardware elements that are critical to safe and reliable system operation. The safety analytical techniques assess design configuration and operational tasks that represent a potential hazard to personnel and equipment. These techniques identify and categorize system hazards by severity and impact. The SAA integrates these reliability and safety techniques to ensure that all critical items and system hazards are identified.

OBJECTIVE	METHOD	KEY ELEMENT
Identify critical items and system hazards.	Perform reliability and safety analyses of the design.	The System Assurance Analysis integrates complementary analytical methods that identify critical items and system hazards.
Track identified critical items and hazards and their associated resolution action to eliminate or minimize risk.	Tabulate identified critical items and system hazards. Maintain summary sheets to reflect status of resolution action.	The Management Information System lists critical items and system hazards, describes resolution. Signature blocks provide visibility and ensure management involvement.
Resolve critical items and system hazards	Eliminate critical items and hazards by design or procedure or accept the associated risk.	The Closed Loop Review System ensures visibility and resolution of critical items and hazards. Those not eliminated by design or controlled by procedure are presented for risk concurrence and acceptance.

Objectives Of The SAA Methodology

Figure 1.

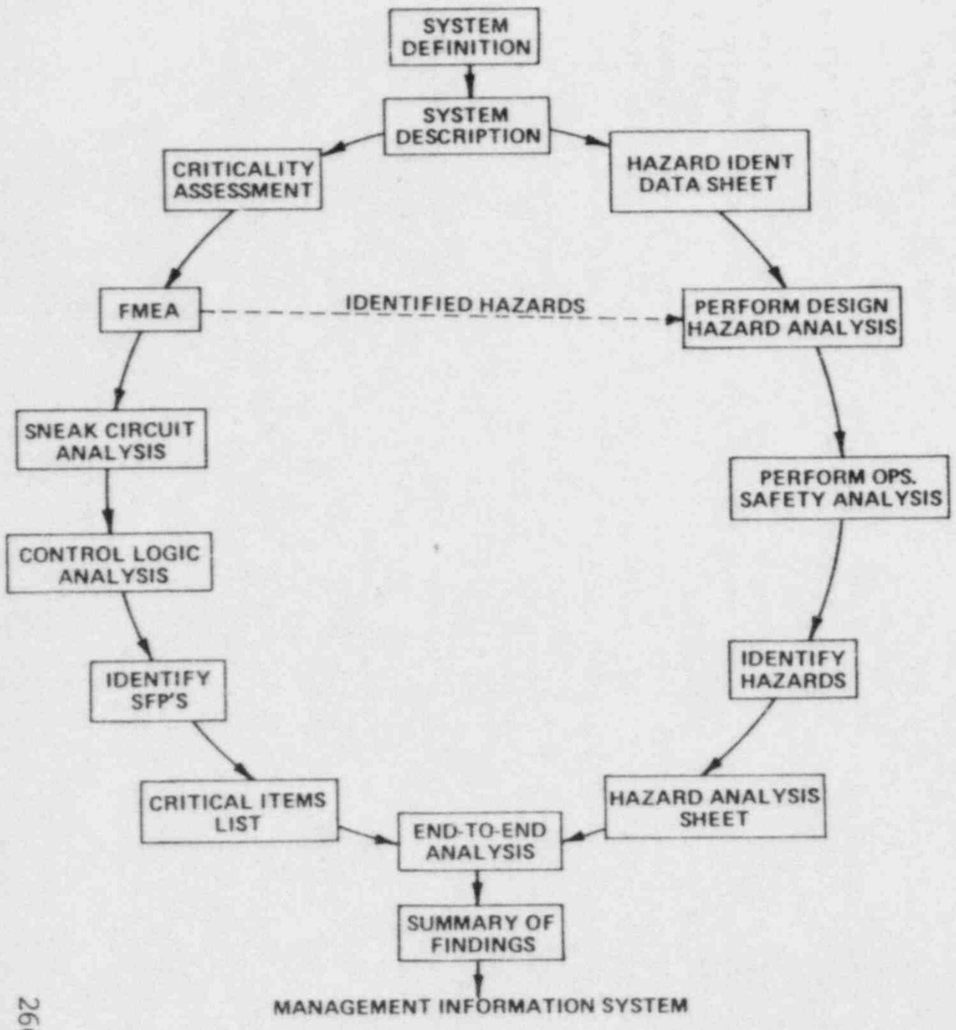

Figure 2
Reliability And Safety Integration

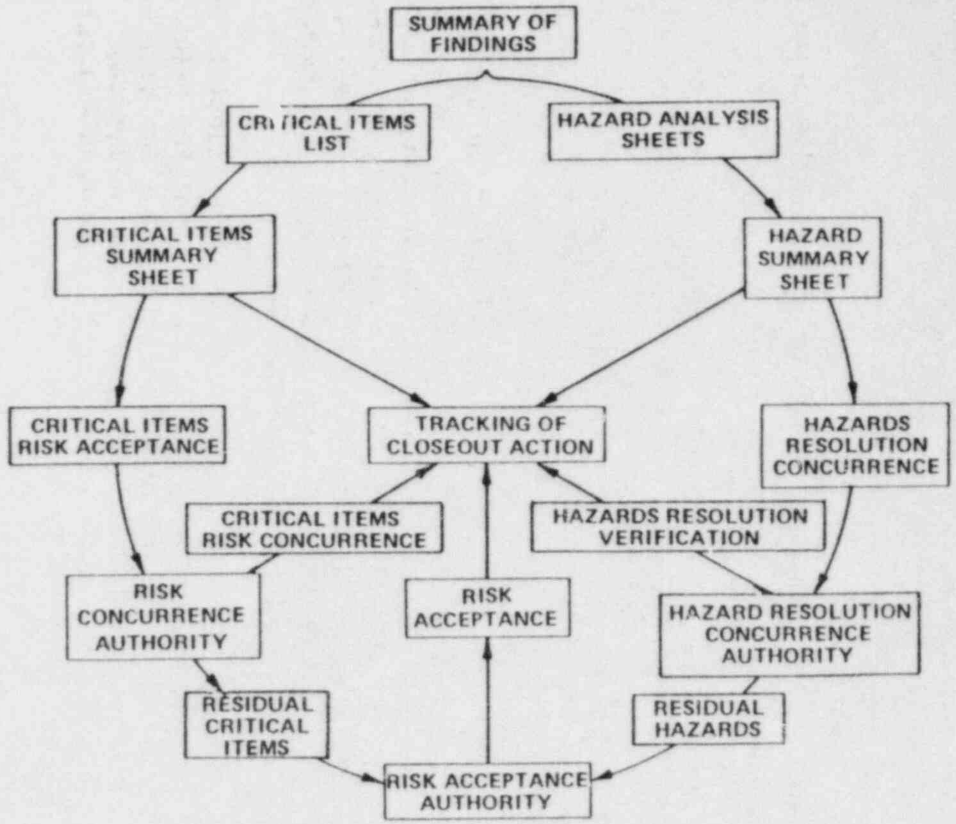

Figure 3
Reliability And Safety Management Information System

The following assessments and analyses are required for completion of an SAA.

a. Criticality Assessment
b. Failure Mode and Effect Analysis
c. Sneak Circuit Analysis
d. Control Logic Analysis
e. Single Failure Point Analysis
f. Design Hazard Analysis
g. Operations Safety Analysis
h. End-To-End Analysis

The Management Information System. The Management Information System is a vital element of the SAA methodology and is initiated and maintained by the analysis team. This system provides management with visibility of critical items and hazards and tracks the status of the resolution of these SAA findings(see figure 3). Inputs to the system come from a summary of the findings of the analyses. These findings are tabulated on the Management Information System Summary Sheets (see figures 4 and 5).

The Closed Loop Review System. The Closed Loop Review System consists of two closed loop review cycles that provide a method to ensure that the resolutions of the results of the SAA are actually implemented into the design process. Figures 6 and 7 show the critical items and hazards review cycles. These review cycles contain both a design review path and a concurrence/acceptance path.

Review Paths The design review path provides a closed-loop path from design, through the review cycle, and back to design. This review path ensures that design modifications, initiated as a result of the review cycle, are fed back into the SAA for assessment. The concurrence/acceptance path provides a closed-loop path from risk assessment, thru risk management, and back to the design review cycle.

Review Package. The review package managed by the critical item review cycle consists of the SAA Critical Items List with attached signature sheet. The review package managed by the hazards review cycle consists of the Hazard Analysis Sheets with attached signature sheet. Signature blocks are provided on the attached signature sheets to identify the applicable review, concurrence, and acceptance authority.

AREAS OF RESPONSIBILITY.

The implementation of the SAA methodology requires the definition of the areas of responsibility for risk resolution, concurrence, and acceptance. Figure 8 shows these areas of responsibility.

SFP SUMMARY SHEET

RELIABILITY & SAFETY MANAGEMENT INFORMATION SYSTEM CRITICAL SINGLE FAILURE POINT (SFP) STATUS

SAA: SAA01FS030-001, REV. B, HYPERGOL, HMF RESP. DE ENGR: J. DOBSON
DATE: 11-24-82
B/L: 42 - 47.01

SFP #	PMN	FIND NO.	CRITICALITY CATEGORY	BOARD ACTION	CONCURRENCE/ACCEPTANCE					
					DE	SF	SP	TS	VO	CD
F01	S70-0868-3D/ S70-0868-4D	A98127	2	ACCEPTED BY CDR	X	X	X	X	X	X
		A98128	2	ACCEPTED BY CDR	X	X	X	X	X	X
		A98129	2	ACCEPTED BY CDR	X	X	X	X	X	X
		A99127	2	ACCEPTED BY CDR	X	X	X	X	X	X
		A99128	2	ACCEPTED BY CDR	X	X	X	X	X	X
		A99129	2	ACCEPTED BY CDR	X	X	X	X	X	X

CRITICALITY	1	1S	2
NUMBER	0	0	6/0

Figure 4

HAZ SUMMARY SHEET

RELIABILITY & SAFETY MANAGEMENT INFORMATION SYSTEM HAZARD STATUS

SAA: 09GS05-001, REV B, ECS, LOA DE ENGR: M/CAPELLIN
B/L: 4

HAZ/ SFP NO.	CRITICALITY CATEGORY	RECOMMENDED RESOLUTION	BOARD ACTION	CONCURRENCE/ACCEPTANCE DE	SF	SP	TS	VO	CD
H01	CONTROLLED		ACCEPTED BY BOARD	X	X	X	X	X	N/R
H02	CONTROLLED		ACCEPTED BY BOARD	X	X	X	X	X	N/R
H03	CONTROLLED		ACCEPTED BY BOARD	X	X	X	X	X	N/R
H04	CONTROLLED		ACCEPTED BY BOARD	X	X	X	X	X	N/R
H05	ELIMINATED	DESIGN CHANGE	SEE SHEET B						

CRITICALITY	CON	RES.CRIT	RES.CAT.
NUMBER	4/0	0	0

Figure 5

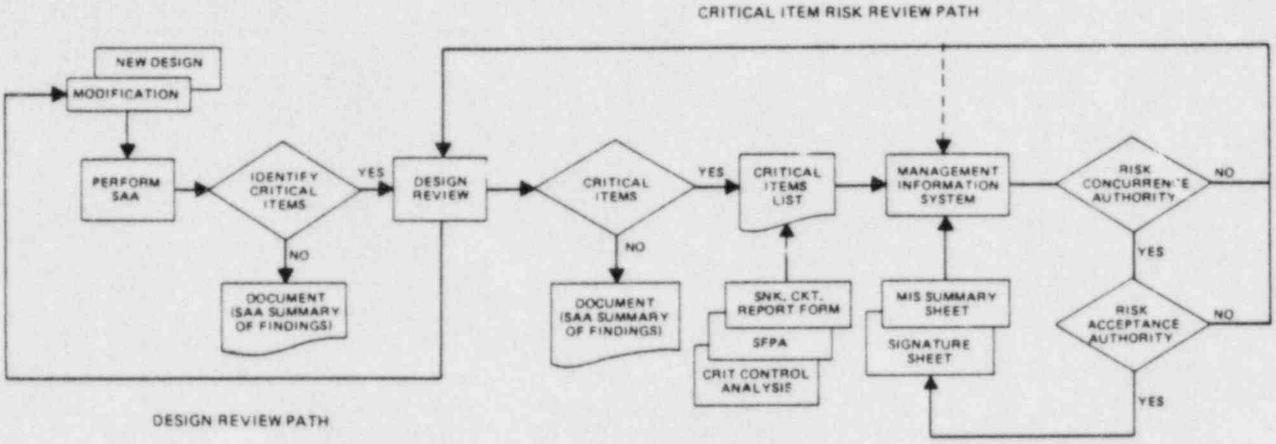

Figure 6.
Critical Items Closed Loop Management Cycle

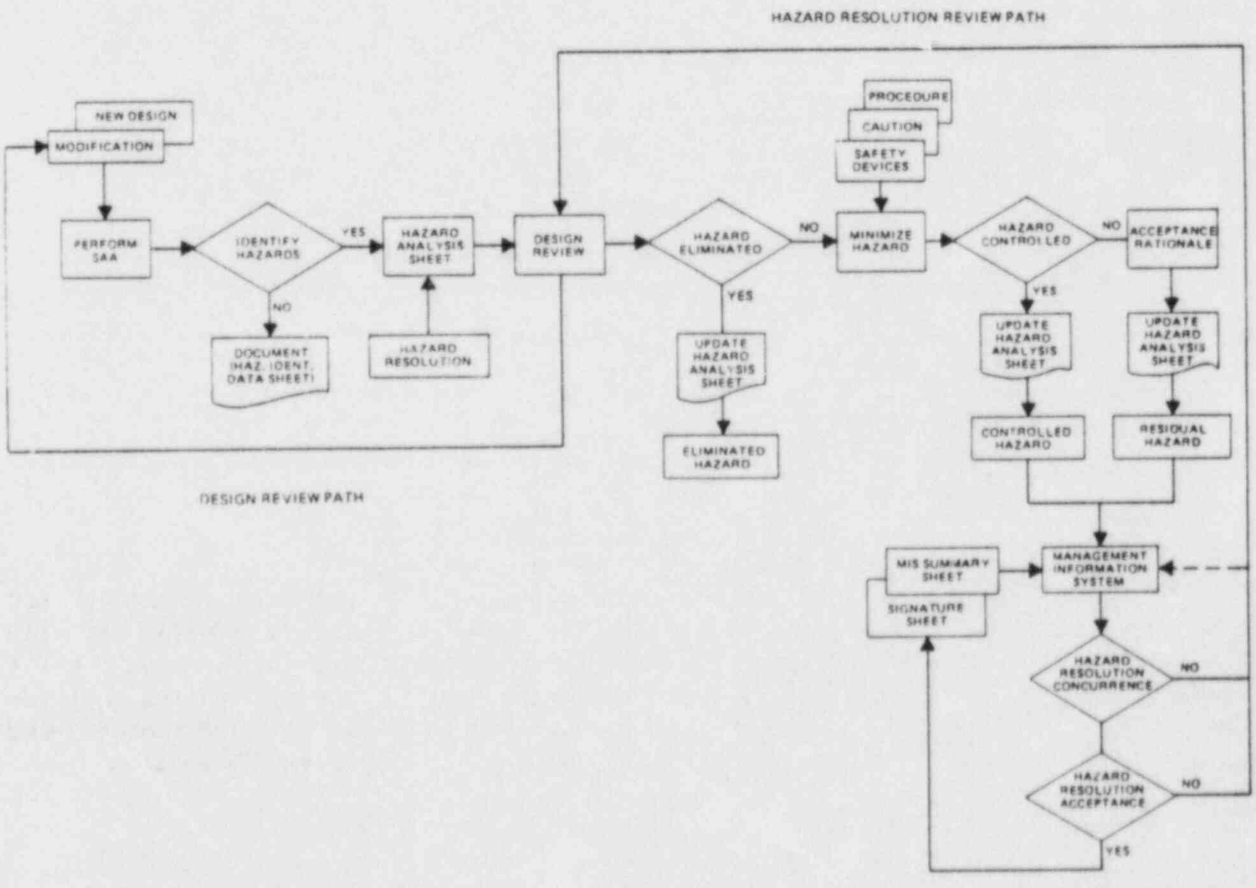

Figure 7
Hazards Closed Loop Management Cycle

NUCLEAR REGULATORY COMMISSION (NRC)	• Establish design and safety criteria • Monitor system assurance activities • Define failure and hazard categories • Identify risk acceptance criteria • Establish risk acceptance authority (residual critical items and hazards)
PLANT MANAGEMENT	• Establish facility baseline/system definition • Organize safety and reliability plan • Implement management elements • Identify risk concurrence criteria • Provide risk concurrence authority
ARCHITECT/ DESIGN ENGINEERING	• Establish document and change control • Conduct design reviews • Provide technical support to SAA • Review SAA results
ENGINEERING SUPPORT (ANALYSIS TEAM)	• Perform SAA (Identify critical items and hazards) • Develop risk acceptance rationale • Develop hazard resolution • Input and maintain information system • Prepare management review package

Areas Of Responsibility

FIGURE 8

IMPLEMENTING THE METHODOLOGY

The implementation of the SAA methodology requires thorough planning for effective management. The operating plant must have a safety and reliability plan that delinates the sequence of system assurance tasks and their relationship to other system life-cycle actions. Planning should include evaluation of other plant functions so that SAA activities complement and support other groups thereby avoiding duplication and overlap of effort.

PLANT MANAGEMENT PREREQUISITE TASKS.

The plant management tasks described in this section are performed to provide planning, develop a core of expertise, and to provide the technical direction and coordination necessary for the success of the analytical effort.

Establish Management Controls. Management controls must be established to ensure the review and subjective application of the results of the SAA. These controls are defined by the elements of the MIS and the CLRS. The management controls are implemented by identifying the concurrence authority and establishing lines of communication between the concurrence authority, the analysis team, and the design organization. Design review milestones are used to implement the design action items required to eliminate identified critical items and hazards.

Establish Concurrence Authorities. The results of the SAA are reviewed by the design organization during the scheduled design review milestones. Critical items and hazards should be eliminated by design changes initiated at the reviews. Those not eliminated by design must be presented to plant management for risk resolution concurrence. Plant Management is responsible for establishing critical item and hazard resolution concurrence authorities.

Critical Item Risk Concurrence Authority. Identified critical items must be reviewed during the design phase for elimination if possible. Those critical items not eliminated by design must be presented to the critical item concurrence authority whose responsibility is to review the items for concurrence of the risk acceptance rationale and to ensure that the accepted resolution is implemented. The critical item concurrence authority is made up of representatives from design, operations, and safety. These critical items are then presented to the risk acceptance authority (NRC) for final acceptance of the risk.

Hazard Resolution Concurrence Authority. Identified hazards must be reviewed during the design phase for elimination if possible. Those hazards not eliminated by design must be reviewed by a hazard resolution concurrence authority. This authority consists of design and operations safety representatives who will review the proposed hazard resolution for concurrence and has responsibility for implementing the resolution action. Residual Hazards are presented to the risk acceptance authority (NRC) for final acceptance of the hazard.

Scope the Analytical Task. The tasks for performing the SAA are multi-disciplined with various skill level requirements. The engineering disciplines and the skill mix required to perform an analysis of a particular system must be identified. Based upon the size of the plant baseline and the complexity of each system, manpower and activity schedules are developed. Support requirements for the team must also be considered when scoping the task.

Organize the Analysis Team. With the correct skill mix, only one analysis team is required to analyze all the systems in the plant. This team should consist of a permanent nucleus of professional personnel with specialized skills in applying scientific and engineering principles and techniques to identify and eliminate or control system deficiencies. System experts would need to support the team only as necessary. The analysis team leader (supervisor) is identified and the analysis tasks are assigned to the team members.

Establish lines of Communication. System experts within the design and operations organizations, for each system defined in the plant baseline, must be identified. Lines of communications between the analysis team and the system experts must be established to provide for a free exchange of information, i.e., technical engineering expertise on the system, operational experience with the system, engineering documentation, preliminary reviews of SAA findings, recommendations for risk acceptance rational and hazards resolutions, etc.

Establish Procedures and Groundrules. The SAA procedures and analyses groundrules must be established to provide clear technical direction and assure continuity within the assessment team. All established procedures and groundrules must be approved by the risk acceptance authority.

TEAM PREREQUISITE TASKS.

The following tasks must be performed by the analysis team prior to beginning the SAA. These tasks are performed to establish the organizational baseline of the facility systems and equipment.

Establish Plant Baseline. The physical organization of the plant must be defined in terms of facilities, systems, and equipment to establish the plant baseline. Establishment of this baseline is necessary in order to identify all of the systems and equipment in the plant.

Define Systems/Equipment. A system can be a single piece of equipment or a grouping of equipment that performs a given function(s). All systems in the plant must be fully defined by appropriate engineering drawings. The engineering drawings must be of sufficient detail so that one could draw a "box" around the system. All interfaces to the system must be identified at the point of interface with the "box". The engineering drawings that define the system become the "model" which is analyzed by the SAA.

The system top document should be a list of all the engineering documents that define that system. Examples of the types of documents required to perform the SAA are: electrical and mechanical schematics, block diagrams, single-line flow diagrams, interconnect drawings, operational procedures, safety and emergency procedures, maintenance and checkout procedures, component detail drawings and specifications, etc.

TEAM ANALYTICAL TASKS.

The definitions and procedures in this section describe the tasks for performing the SAA.

Failure and Hazard Category Definitions. Failure categories are defined in terms of system and system hardware failure effects. This definition allows the assignment of these categories to systems and system hardware based on the severity of their failure effect. Hazard categories are defined in terms of the subjective resolution of the identified hazard. Failure and hazard categories are listed in order of degree of relative severity and allow for positive definition of critical systems and system hardware and provide for prioritizing of hazard resolutions.

Component Failure Categories. The categories which describe the effects of a component failure on a system are as follows:

CATEGORY	EFFECT ON SYSTEM
A	Loss of System - the complete loss of the function of the system.
B	Degradation of System - inability of the system to function at adequate capacity.
C	Loss of Redundancy - the loss of a back-up subsystem(s)
D	No Significant Effect - the system can perform its function at adequate capacity.

System Failure Categories. The categories which describe the effects of a system failure on plant operation are as follows:

CATEGORY	EFFECT ON PLANT
1	Release of Radiation - any failure of a system which could permit the uncontrolled release of radiation.
1S	Loss of a Safety or Hazard Monitoring System - a failure in a system that prevents that system from detecting or combating a Category 1 system failure effect on the plant.
2	Loss of Life or Personnel Injury - a system failure resulting in the potential loss of life or personnel injury.
3	Plant Interruption- the failure of a system or loss of a redundant subsystem that would not cause personnel injury but could cause degraded plant operation, such as plant trip, reduced power, and unit off line.
4	No Effect - no significant effect on plant operation

Hazard Categories. Hazards are categorized by how they are to be resolved. Hazards caused by environment, personnel error, design characteristics, procedural deficiencies, or some equipment malfunctions, that may result in loss of personnel capability or loss of system function, are categorized as follows:

a. Eliminated Hazard - An identified hazard that is to be counteracted by a design change that will eliminate the hazardous condition.

b. Controlled Hazard - An identified hazard that is to be counteracted by appropriate design, safety devices, alarms, caution and/or warning devices, or special automatic or manual procedures, which are required to control the hazard.

c. Residual Hazard - An identified hazard that is not or cannot be eliminated or controlled. The hazard will be assessed and the risk accepted by management. A residual hazard will be classified as one of the following:

   (1) Catastrophic Hazard - no time for corrective action.

   (2) Critical Hazard - may be counteracted by emergency action performed in a timely manner.

Develop System Functional Description. A narrative functional description is prepared for each system that is identified during the systems definition task. The amount of detail is governed by the complexity of the functions performed or by the application of the output of the system. The analyst then prepares a system block diagram that shows graphically all system elements to be analyzed. All system inputs and outputs are shown, enabling assessment and categorizing of the effects of loss of any input or output.

Perform Criticality Assessment. A Criticality Assessment is performed on all of the systems and equipment identified in the plant baseline and reported on a Criticality Assessment Summary Sheet. Using the functional descriptions developed above, the functions of the system are assessed and assigned a system failure category. Systems whose loss of function or improper functioning could result in a category 1, 1S, or 2 system failure effect on plant operations are defined as critical systems. Systems whose loss of function or improper functioning could result in a category 3 system failure effect on plant operations are defined as crucial systems. The Criticality Assessment examines the functions of the system and establishes the criticality of the system by assessing the failure effect of each function. A critical system does not necessarily contain any critical elements, such elements may already have been eliminated by design redundancy. Criticality is based on failure effect due to loss of function, not on design. The criticality category of the system is defined as the highest failure category of any of it's functions. The Criticality Assessment establishes the level of review of the original design as well as establishing the level of review for all future changes or modifications.

Perform Failure Mode and Effects Analysis (FMEA). Systems and equipment that are assessed as critical in the Criticality Assessment will be subjected to an FMEA and documented on FMEA worksheets. The FMEA is performed to identify system hardware elements that represent a potential Critical Single Failure Point (CSFP). A CSFP is any system component whose failure mode(s) results in a category 1, 1S or 2 failure effect on plant operation. The FMEA provides the following data:

a. A list of all system components considered active and their associated failure modes

b. Concise statements regarding the failure modes and failure effects on system operation for each component addressed

c. Assignment of a component failure category to each component failure mode in accordance with the Failure Effect on System Operation categories

d. Concise statements regarding the failure modes and failure effects on plant operation for each component addressed

e. Assignment of a system failure category to each component failure mode in accordance with the Failure Effect on Plant Operation categories

Perform Sneak Circuit Analysis. The Sneak Circuit Analysis, performed on all electrical/electronic circuits in critical systems, is an analytical approach to detection of latent conditions that could or would cause unwanted functions to occur or inhibit wanted functions independent of component failure. The technique involves accumulation of detailed circuit diagrams and wire lists, arrangement of circuit elements into topological network trees, and examination of these trees for susceptibility to sneak circuits. All critical sneak circuit conditions will be reported on the Sneak Circuit Report Form and listed on the CIL.

Perform Control Logic Analysis. A Control Logic Analysis is performed on all critical systems. System command, control, and response functions are examined to identify unintended sequences that can result in undesirable operations or can inhibit desired functions. The system electrical schematic is reviewed and control discrete and analog functions are identified. The analysis begins with grouping control functions, related by common initiating hardware, and assessing the effect of loss of all commands. Discussions concerning the function, control, and instrumentation affected by each control function are prepared. Control functions assessed as critical are described on the Critical Control Function Analysis Sheet.

Perform Single Failure Point Analysis (SFPA). A component, assessed on the FMEA worksheet, whose failure effect on plant operation falls into a critical category, is defined as a Critical Single Failure Point (CSFP). Each CSFP that is not eliminated by the design review process is analyzed in detail on a SFPA worksheet. The SFPA worksheet repeats the information listed on the FMEA worksheet and adds risk acceptance rationale. The acceptance rationale consists of design versus use parameters, test and inspection information, and failure history. The failure history includes figures of reliability in addition to any history of failure, especially in the critical failure mode.

Perform Design Hazard Analysis. The Design Hazard Analysis is performed on all systems in the plant baseline regardless of system redundancy or criticality category. This analysis is an analytical technique for evaluating a system design to identify potential hazards to personnel and equipment and develop the resolution necessary to eliminate or counteract the identified hazards. This analysis complements the FMEA by analyzing hazardous situations normally groundruled out of the FMEA, such as hazards caused by human error, failure of passive components, or multiple/common mode component failures. Identified design hazards are assessed on a Hazard Analysis Sheet.

Perform Operations Safety Analysis. An Operations Safety Analysis is performed on all human and machine tasks. This analysis identifies tasks that are inherently hazardous or, by the nature of the task, can lead to the development of hazards in the operation of the system. This analysis is performed to identify and resolve hazards existing or are developed during a particular task. Such hazards may result from the task itself, or from interaction of other tasks being done concurrently. Identified operational hazards are assessed on a Hazards Analysis Sheet.

Perform End-To-End Analysis. The End-to-End Analysis considers all the other systems which function as part of, or may impact, the on-line operation of the system being analyzed regardless of design responsibility or system interface. This analysis identifies areas of concern that are outputs of interfacing systems or support systems that may degrade operation of the system under investigation. Identified areas of concern are documented as either a critical item or a design hazard.

Critical Items List (CIL). The CIL is prepared for each critical system and is a listing of all CSFP's, critical control elements, and sneak circuit conditions, and the associated SFPA worksheets, Control Function Analysis worksheets, and Sneak Circuit Report forms. The CIL is used to inform management of the risks associated in operating a particular system. The information in the CIL is also used to assist management in determining whether additional resources should be committed to eliminate the critical items or if the risk can be accepted and left in the system. Each item on the CIL is reviewed based on its own risk. A set of prior approved groundrules for acceptance of risks must be established by the risk acceptance authority to insure that the system meets program safety and reliability goals.

DISPOSITION OF CRITICAL ITEMS.

Critical items are tracked and annotated on the Management Information System summary sheets as OPEN while they are in the review cycle. The action required by the acceptance rationale to mitigate the associated risk for those items not eliminated by design must be accepted by the Risk Acceptance Authority. These critical items remain OPEN until the Risk Concurrence Authority ensures that the approved actions are implemented. Then the critical items are considered CLOSED. All OPEN critical items are a constraint on the operation of the system.

DISPOSITION OF HAZARDS.

All identified hazards are tracked and annotated on the Management Information System summary sheets as OPEN while they are in the review cycle. Eliminated or Controlled hazards remain OPEN until eliminated by design or controlled by procedure. Residual Hazards that can neither be eliminated by design nor effectively controlled by procedure are never closed. The risk must be accepted by the risk acceptance authority and periodically reviewed at appropriate design and operational milestones. Specifically, all identified hazards are categorized and tracked as follows:

a. Eliminated Hazards. Hazards of this category remain open until a design change has been accomplished that eliminates the hazard. When the design change has been physically accomplished, the hazard is considered a closed eliminated hazard.

b. Controlled Hazards. Hazards of this category cannot be eliminated, but the risk can be reduced to an acceptable level by design change or imposition of a safety procedure. When the change or procedure is in effect, the hazard is considered a closed controlled hazard.

c. Residual Hazards. Hazards of this category can neither be eliminated nor controlled and must be accepted by the risk acceptance authority. Residual hazards are never closed. Since subsequent design or procedural changes could change the category to controlled or eliminated, residual hazards should continually be addressed within the design change process.

CONCLUSIONS AND OBSERVATIONS

Nuclear power plant systems and aerospace ground support systems are similar in complexity and design and share common safety and reliability goals. The SAA methodology is readily adaptable to nuclear power plant designs because of it's practical application of existing and well known safety and reliability analytical techniques tied to an effective management information system. An SAA started early in the design engineering process becomes the foundation for an expanded analysis as design progresses. Tracking of the results of the analyses continues after design is complete and operations and test procedures are defined. This methodology isn't finished just because the design has been completed and the systems operational. The results of the analyses are tracked and managed as long as the system is operational.

The Chief Counsel's Report to the President's Commission on the Accident at Three Mile Island identified many concerns related to the treatment of unresolved safety problems (residual hazards). Incorporation of the SAA methodology into the design review process would provide a method for resolving these concerns.

The SAA provides a systematic method of identification and resolution of critical items and hazards found during the design process. When the design reaches the 100% review level, the acceptance rationale for remaining critical items and the resolution of residual hazards are presented to the reviewing agency for evaluation of the risks involved and acceptability of the design. This methodology documents the safety and reliability problems identified during the design process and provides management with a focal point where these problems can be assessed.

While the feasibility of the SAA methodology has been demonstrated, some practical considerations need further exploration. First, for this methodology to be most useful and attractive to the nuclear power industry, it should be an integral part of the design and/or licensing process, not just an additional requirement. Changes in existing regulations and industry practices would be required before the SAA methodology could replace current requirements. Second, the present direction taken by the NRC is to establish safety goals with numerical guidelines. The SAA methodology does consider the probability of failure in the single failure point assessment, but this consideration is only one of the many items examined to determine if a single failure point is an acceptable risk. Many other factors are also considered such as: design safety margins, qualification testing, mandatory inspections, training and certification requirements, operating procedures, and other related items.

THE APPLICATION OF RELIABILITY-BASED TECHNIQUES TO TECHNICAL SPECIFICATIONS

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1. INTRODUCTION

With the Nuclear Regulatory Commission (NRC) beginning to focus on questions of long term nuclear power plant operability, and as the techniques and related data utilized in risk and reliability analysis continue to mature, we have seen an increased use of probabilistic safety assessments. Currently risk and reliability approaches have been incorporated into the evaluation process of various NRC offices: Inspection and Enforcement (I&E), the Regions, and Nuclear Reactor Regulation (NRR). Programs are ongoing to use the Limerick power plant's Probabilistic Risk Assessment (PRA) and its related Severe Accident Risk Assessment to prioritize preoperation and start-up inspection requirements and to aid in the actual audit of these functions at the site. A similar effort is underway to prioritize the inspection needs at the Indian Point site, Units 2 and 3. The use of probabilistic analyses has also been seen in resolving the questions of the siting of Indian Point 2 and 3 and Zion 1 and 2 near high population densities. When Big Rock Point evaluated the feasibility of meeting all the TMI action plan items, the NRC staff and the licensees used the plant's PRA as a source of information on cost effectiveness of the retrofits. PRA has also been applied to respond to the station blackout, anticipated transient without scram (ATWS), and containment sump/recirculation performance issues.

As the "safe operation" issue is further pursued, questions of the adequacy of test and maintenance practices and the minimum required equipment list for safe operations become of increasing importance. The review of the Reactor Protection System (RPS) testing needs, after the Salem failure to trip incident, was handled in part through a time-dependent reliability code called FRANTIC. A similar review of the RPS limiting conditions of operation was also conducted through the application of PRA tools.

All in all, the use of risk and reliability methods in the analysis of the effects of test, maintenance, and surveillance on long-term nuclear power plant operational safety is a natural extension of current successful applications and follows the efforts discussed in NUREG/CR-3627 and NUREG-1024 and the proposed NRC "Maintenance and Surveillance Program Plan" dated July 31, 1984. The remaining sections of this paper address the Probabilistic Evaluation of Technical Specifications (PETS) program currently underway at Brookhaven National Laboratory to address this issue.

2. DISCUSSION

2.1 Background

The requirement for NRC license applicants to state Technical Specifications (TSs) related to its facility, for which an operating license is being requested, has been recognized from the initial passage of the Atomic Energy

Act in 1954. Although all parties agree that the NRC has the responsibility in requesting TS input, participants in the NRC licensing reviews have long disagreed on which items should be included as TSs, since no specific ruling exists which provides a selection criteria. The absence of specific criteria has resulted in numerous items of vastly differing levels of importance being included and specification requirements that are inconsistent and which, if complied with, may be adverse to plant safety. Also, there is no coherent basis through which the nuclear industry can justify temporary exemptions of particular requirements within the TS, nor which the regulator can use for granting such exemptions.

Indeed, Technical Specifications are a crucial part of the regulatory fabric. As required by 10 CFR 50.36, plant Technical Specifications (TSs) for power reactors are to include (1) safety limits and limiting safety system settings, (2) Limiting Conditions of Operation (LCO), (3) Surveillance Requirements (SRs), (4) design features, and (5) administrative controls. Purportedly, to make the TS process (which applicants for operating license are required to utilize) more effective and efficient, the NRC developed and required, on a forward-fit basis, the use of Standard Technical Specifications (STs) since 1975.

In part, and in attempting to comply with the elements contained within TSs, usually engineering judgment, codes and standards requirements, and manufacturer's recommendations are used as the basis for establishing testing and surveillance policies specified in the SR element of TSs. However, situations have arisen which indicate that test and surveillance intervals that are either too short or too long could, through different mechanisms, be adverse to safety.

A recent Executive Director's Office (EDO) Task Group report on enhancing the safety impact of plant TSs (NUREG-1024) has resulted in a directive wherein NRR/DL has been given the lead to develop and implement a program that would accomplish the intent of those recommendations put forth by the Task Group. Implementing these recommendations could reduce any adverse impact that might result from complying with existing STs.

Basically, the findings of the EDO Task Group are that although engineering judgment must still be the primary basis for establishing, overall, each of the elements which comprise TSs, insights gained from probabilistic methodologies can be a significant aid in arriving at and in rationalizing these judgments. These probabilistic methodologies could be used in the decision making process for establishing component/system test and maintenance policies. Downtime extensions, i.e., extensions to Allowed Outage Times (AOTs), testing intervals, and testing procedures can be investigated using these methods, and decisions can be made (when compared to one or several risk-measures) as to how TSs may be modified or presently evaluated in the context of plant safety.

In this connection, BNL was requested in February 1984 by RES to examine approaches for developing a quantitative basis for making engineering judgments in revising the STSs. The members of the BNL program task force responsible for the work reported herein are: P. Samanta, S. Wong, J. Fragola, E. Lofgren, and W. Vesely. The work reported on in this paper is a result of this team and is in part extracted from current BNL reports as indicated.

2.2 Project Overview

To scope out the PETS program, designed to address the safety implications and to streamline the regulatory process in complying with either TSs or STSs, requires reconciliation of or attention to some rather broad issues. Only when these broad issues have been addressed, or at least clearly identified, can one focus on the more narrow perspective of testing and maintenance policies addressed within the technical specification.

Several issues arise in addressing various alternative means for evaluating the safety implications of technical specifications. In a hierarchical order the following must be addressed:

What are the possible approaches for evaluating technical specifications?

What are the possible measures of technical specification performance?

What technical specifications aspects (attributes) need to be evaluated?

What are the possible objectives of technical specifications?

Figure 2.1, depicts this hierarchy from a reliability viewpoint in more detail. The possible safety objectives of TS should be to control risk, monitor and assure reliability. This is accomplished by evaluating certain attributes such as testing, downtimes, maintenance, etc., as shown in the third level of Figure 2.1. In turn there are at least four (4) possible measures that can be utilized to investigate TS performance. These are public risk, occupational risk, reliability impact and costs, each of which have various evaluation approaches, that can be applied (reference Level 5). Under FIN A-3230, Time Dependent Reliability Modeling, BNL has, over the past several years, been developing, improving, and extending mathematical models (FRANTIC) to address the first three elements of Level 3, Testing, Downtimes, and Maintenance. Under FIN A-3231, a project that had been cancelled, models to address LCOs and SR were being developed. These two programs were addressing system unavailability due to test intervals and allowed outage times. In addition to these RES-sponsored programs, NPR-sponsored programs, at BNL, have also studied the risk-implications of test interval and allowed outage times using a Markov system transient model. This experience has also been used in drafting this program plan and is a requisite ingredient in its implementation.

However, as a subset for the overall evaluation of TS, the analysis of alternatives for test interval and down time extensions, requires additional efforts. These are schematized in Figure 2.2. Figure 2.2 displays the hierarchy of these alternatives by asking the following questions:

What are the bases for the evaluation?

What type of criteria should be used to determine acceptability?

What specific attributes need to be considered?

What plant models should be used?

What techniques and codes can be used?

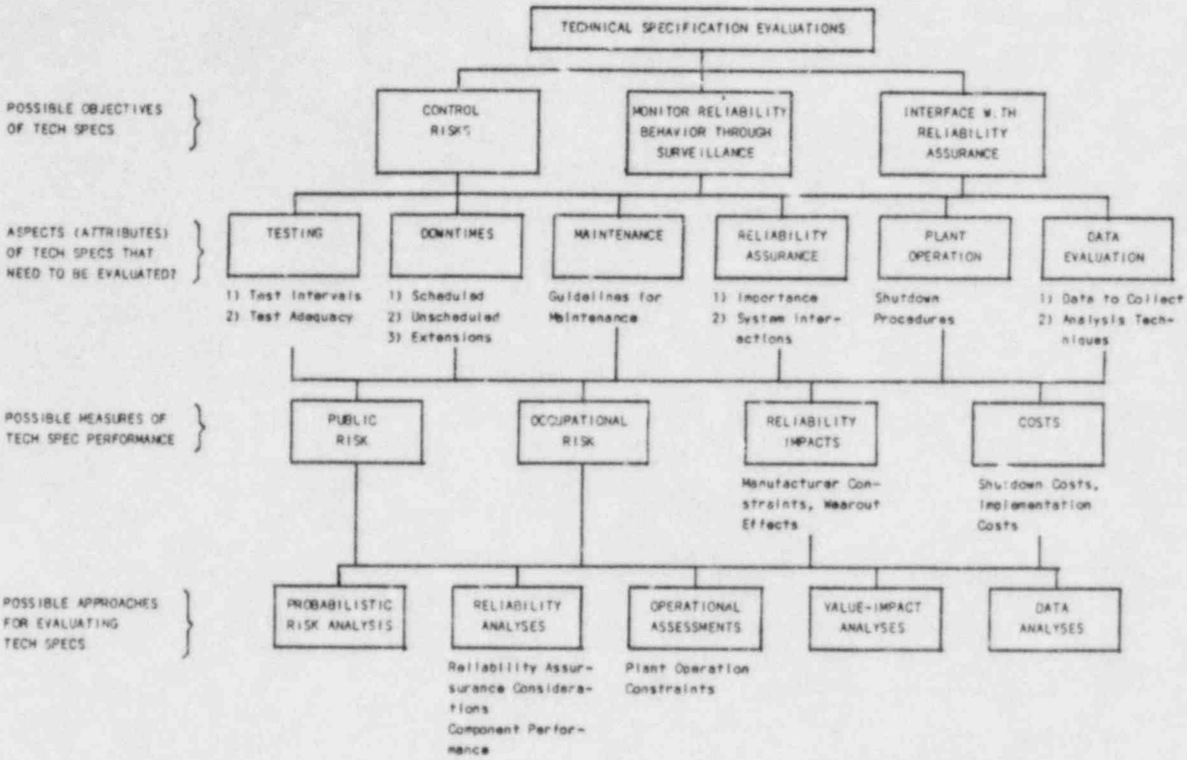

Figure 2.1 Hierarchy of alternatives for Technical Specifications evaluations

Figure 2.2 Hierarchy of alternatives for test interval and downtime evaluations.

The TSs, being the governing requirements by which a nuclear power station is operated, can have a direct and significant impact on safety. Therefore, the research program proposed must interface with other ongoing NRC projects. The PETS program will, by necessity, depend on the output of these other projects and will also be in a position to supply programmatic guidance to them. An example of interfaces requiring integration that have at this time been identified, are shown in Table 2.1. In addition to supplying its own output on TSs, PETS must also be the "road map" that can be used by other projects for better understanding of and providing assurances to safe operations.

Table 2.1 Interfaces with Other NRC Projects

1.	Operating Safety Reliability Research (OSRR)	Specific items for reliability assurance and reliability monitoring will be identified in this program and will be passed to this effort. This program will not define the reliability assurance and reliability monitoring procedures which can be implemented since this is an OSRR function.
2.	Accident Sequence Evaluation Program (ASEP)	It is anticipated that PETS will utilize the ASEP system models, accident sequence models, and data as a basis for the evaluations which will be performed. Modifications will be identified which are required in the ASEP models for effective utilization of the models in evaluating technical specifications. Major ASEP model modifications will not be performed in this program since this is not an objective of the program.
3.	IE and Regional Prioritization Program	This interface is principally one of communication to ensure consistency between the prioritization programs and the prioritizations implied by technical specification recommendations developed for this program.
4.	Inplant Reliability Data System (IPRDS)	Data needs will be identified which will be passed to the IPRDS program to assist in developing data objectives and priorities.
5.	Maintenance & Surveillance Program	This human factors effort will utilize PETS to establish the systematic needs of maintenance intervals, outage times, and surveillance intervals. In turn the output of the maintenance and surveillance program can be tested for risk reduction through the use of the output of PETS.
6.	Technology Transfer Program	PETS will supply guidance to this program to anchor it to real NRR and I&E decision needs. PETS will also act as a catalyst in the development of new applied course material.

2.3 Programmatic Framework

The five areas covered by the plant TSs, as described in Section 2.1, are displayed pictorially in Figure 2.3. The scope of the PETS program will concentrate on two of these, namely the LCO and the SR. These areas readily lend themselves to analysis and enhancement by reliability techniques and in general include the nine plant functions that should be addressed. Current TSs and, in particular, the LCOs and SRs, have been established primarily by using engineering judgment, manufacturer requirements and industry codes and standards. What is proposed here is that a third dimension be added to TSs. This dimension is that of reliability insights. In this manner the current basis, where valid, can be kept and simply augmented by reliability analysis. The PETS program will not only implement the analytical tools and reliability insights needed to evaluate TS, but will also integrate them. Since the program touches upon many technical areas, the plan, as described in this report, must be considered as a dynamic framework. As PETS matures and as other NRC and industry projects move forward, the approach described here should be reevaluated and where necessary fine tuned. In addition, this framework should allow for the consideration that the recommendations developed be implemented in a phased way with early results. More engineering-oriented, experienced-based, approaches characteristic of reliability research program, can then follow.

While the development of an approach to address alternatives for test interval and downtime evaluations will be important, the breadth and depth of the larger problem, TS evaluations, requires that the study be implemented in structured, systematic manner which properly considers all the interfacing programs both within and outside the NRC, and allows for the development of approaches while covering a broad spectrum from stand-alone engineering judgment to the sophisticated analytical models.

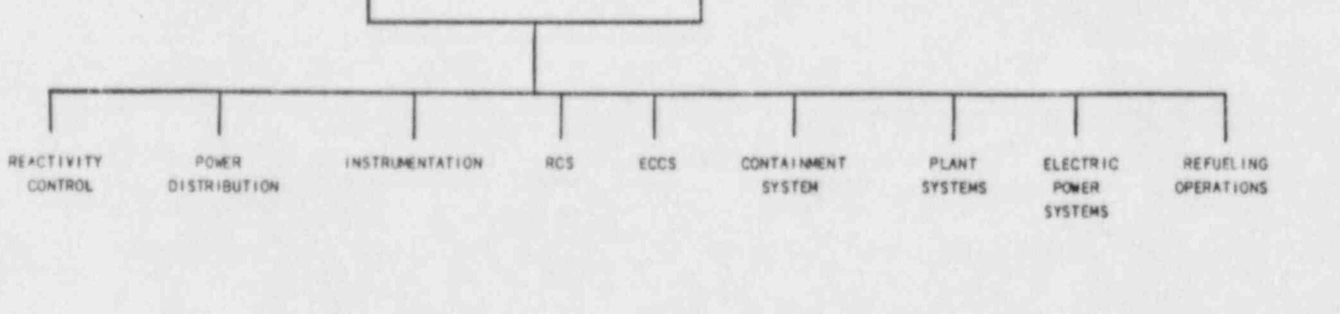

Figure 2.3 Technical Specifications: Illustrative sections of interest.

3. THE PETS PROGRAM

The PETS program, initiated in February 1984, was structured to first develop a detailed program plan defining the short-term (15-month) and long-term products. After extensive peer review, seven tasks were defined (see Figure 3.1). With the final program plan having been approved, Tasks 2 through 6 are currently well underway, and Tasks 5 and 6 have produced draft reports. The remaining subsections of this section discuss each task. For a more detailed discussion, the reader is encouraged to review the PETS final program plan dated October 1984. A discussion of the interim results from Tasks 5 and 6 will be presented in Section 4 of this paper.

Task 1 - Development of Program Plan
Task 2 - Plant Model Development
Task 3 - Data Base Development
Task 4 - Model and Software Development
Task 5 - Determinations of AOTs
Task 6 - Determinations of TIs
Task 7 - Implementation Program Development

Figure 3.1

3.1 Task Outline

Task 1 - DEVELOPMENT OF PROGRAM PLAN

The objective of this task was to develop a detailed program plan which will be used as a guide throughout the PETS program. The program plan identifies the various issues involved in TS evaluation, defines the interfaces with other NRC and industry sponsored programs, and stresses the development of products directly usable by the NRC and the industry.

The PETS program has begun to develop interfaces with other related activities. They include the owners group programs on technical specifications, EPRI-sponsored activities, and the NRC OSRR Project. This program also depends on other programs, like ASEP and IPRDS, for supporting information. The program interfaces with the requirement from other programs and the necessity of conducting peer review to coordinate the impact of various related activities are also included in the program plan.

The final version of the plan, incorporating the comments received from the peer review of the two draft program plans, is the major product from this task and is completed. However, the final program plan will be a dynamic document and will be modified, if found necessary, based on the progress in and the insights gained from the other tasks.

Task 2 - PLANT MODEL DEVELOPMENT

The objective of this task is to select plant models having available system models and accident sequences for subsequent application of various approaches for evaluating TSs and for demonstrating the implementation of alternative procedures. The selection of plants, system models and accident sequences are being conducted with the end objective of recommending those plant models needed in the TS investigation. In that sense, this task will continue at various stages of the project with sufficient flexibility to address the needs of the other tasks, to accommodate the input from other on-going system model and accident sequence development programs as well as to keep "in-step" with concomitant and concurrent efforts by industry.

This task consists of four subtasks: a) the selection of plants, b) the selection and development of system models, c) the selection and development of accident sequence models, and d) the selection of STSs and plant operating procedures. Each of these subtasks are described below.

To meet the time constraint, the plants selected must have system fault trees and accident sequences already developed. To address the question of the level of detail necessary for conducting a TS evaluation, the plants chosen should also have had a detailed PRA performed on them and be included in the ASEP models for comparative purposes.

In addition, for those systems fault trees selected, cut sets at the component level must have already been determined and all hard-wired interfaces and major risk contributors have been identified. The accident sequences selected will be the dominant accident sequences containing the systems selected. The initial analysis will be conducted on the basis of selected ASEP accident sequences. However, ASEP models do not contain all systems for which it is necessary to evaluate or establish AOTs and SRs. This task is not expected to develop any system models or accident sequences; it is anticipated that ASEP will be expanded to include these additional systems and sequences, such as those involving small LOCA initiating events, low pressure system failures, long term cooling failures, and reactor protection system.

Within this task all the information relevant to the selected system models and accident sequences is being reviewed and assembled; the fault trees and minimal cut sets are being prepared to be in a format usable by the software packages.

In addition, this task requires assembling the TS procedures and plant operating procedures for those components to be evaluated. This includes currently allowed maximum downtimes and testing intervals. Other information such as type of tests performed, type of repair and maintenance performed, and system reconfiguration during test and maintenance are being collected. This effort is currently ongoing.

Task 3 - DATA BASE DEVELOPMENT

The objectives of this task are essentially to identify the data base requirement for TS evaluation and to generate the data base in terms of best estimate and ranges (bounding values). In addition, a data base relating to specific needs of this program must be developed through which recommendations on the need of a generic data base for subsequent use in TS evaluation can be made.

Task 3 consists of three major subtasks. First is the development of an interim data base to support the plant models (ASEP, etc.); second, is the additional data needed with component specific failure and repair models, and third, is the development of a guidance document intended to supply utility licensees with instruction as to the construction, contents, and structure of a living data base which is considered as acceptable for use in support of petitions of TS relief. A final data program plan was issued under a modified scope of work and designated as a parallel effort to this task. The work on the data program plan is completed. The actual data base work is currently ongoing and will be submitted as a draft interim report in October 1984.

Task 4 - MODEL AND SOFTWARE DEVELOPMENT

The objectives of this task are to assemble and review available analytical models applicable for TS evaluation to determine their usefulness in the program. Also, available software packages will be assembled and reviewed to determine their applicability. As part of these reviews, areas where further model development may be necessary will be identified and modifications of the models and software packages that are deemed necessary for carrying out the application and demonstration of TS methodology will be made.

The models include the standard PRA techniques, time-dependent models like FRANTIC, MARKOV models, allowed downtime and surveillance interval models being developed at BCL and SAI, and other analytical reliability models that have been developed. The inherent capability of each of the models to address aspects of the problem will be reviewed and determinations will be made where the models, with modifications, can address additional issues. Test runs will be made to determine the level of detail actually required, the adequacy of models, and their relative strengths for use in TS evaluation. Particularly, the advantage of using detailed time-dependent as opposed to time-independent models will also be determined on the basis of selected evaluations. This effort is ongoing.

Task 5 - DETERMINATIONS OF ALLOWED OUTAGE TIMES

This task has the following multiple objectives:

- To develop measures to evaluate the risks and benefits associated with AOTs.
- To develop criteria by which to judge the acceptability of AOTs.
- To address issues associated with AOT determination.
- To specifically address the interactions between the determination of AOTs and determination of SRs.
- To develop specific procedures for determining AOTs.
- To develop specific procedures for determining AOT extensions.
- To test and demonstrate procedures on specific systems and sequences.

Task 5 is developing specific procedures for determining acceptable AOTs and acceptable AOT extensions. A comprehensive selection of alternative approaches is being developed for determining AOTs and these various approaches will be evaluated for their benefits and limitations. A first draft report has been sent to the NRC staff for review. The specific areas which are addressed include:

- The use of different absolute and relative risk measures in determining AOTs.
- The ramifications of determining AOTs on a system level, accident sequence level, core melt frequency level and general risk level.
- The role of value-impact analyses including the consideration of occupational risks and shutdown risks.
- The impact of multiple failures and the effect of test before repair.
- The role of safety goals and allocation
- The role of prioritization and importances.
- The effect of variations on the occurrence frequency of downtimes.
- The characteristics of a cumulative downtime approach.
- The effects of human error consideration and human recovery consideration.
- The effect of common cause considerations.
- The role of failure cause and repair completion time knowledge.
- The evaluation between AOT determination and TI determination.
- The effect of uncertainty considerations.

Task 6 - DETERMINATIONS OF TEST INTERVALS

This task has the following multiple objectives:

- To develop measures to evaluate the risks and benefits associated with TIs.
- To develop criteria by which to judge the acceptability of TIs.
- To address issues associated with TI determination.
- To specifically address the interactions between determination of AOTs and determination of TIs.
- To develop specific procedures for determining TIs.
- To test and demonstrate procedures on specific systems and sequences.

Task 6 is developing specific procedures for determining acceptable TIs. A comprehensive selection of alternative approaches is being developed for determining acceptable TIs and these alternative approaches will be evaluated for their benefits and limitations. A first draft report has been issued for comment on this task. The specific areas which will be addressed will include:

- The use of different absolute and relative risk measures in determining acceptable TIs.
- The ramifications of determining TIs on a system level, accident sequence level, core melt frequency level, and general risk level.
- The role of value impact analyses including test cost considerations and occupational risk considerations.
- The impact of deleterious effects of testing.
- The impact of different testing schemes.
- The role of safety goals and allocation.
- The effects of human error considerations and human recovery considerations.
- The effect of common cause considerations.
- The effect of wear-out considerations.
- The interactions between AOT determinations and TI determination.
- The effect of inefficient or ineffective testing.
- The effect of manufacture and warranty constraints.
- The effect of uncertainty considerations.

From the thorough review of the candidate approaches developed, and from the comprehensive applications that will be performed, specific procedures will be identified for determining acceptable TIs which are robust to uncertainties and which are implementable. These procedures will address determining TIs based on a generic level and plant specific level. The application that will be carried out will not only demonstrate and test the approaches but will be used as a basis for proposing specific TIs for those cases studied.

Task 7 - IMPLEMENTATION PROGRAM DEVELOPMENT

The objective of this task is to develop an implementation program based on the guidelines generated in the previous tasks. The implementation program will be developed for each aspect of the TSs to be evaluated. They include determination of AOTs and determination of TIs.

For each aspect of the TS evaluated, an implementation program will be de veloped based on the guidelines developed within the previous tasks. This implementation program will include a logic flow diagram for using the guidelines recommended computer codes and types of analyses to be performed, recommended level and detail of plant models and the data base to be used for the evaluation. This will also include the issues that must be treated and recommended ways of treating those issues.

3.2 Program Products

The objective of the PETS program is to produce products which are directly usable by the various offices of the NRC and the industry for reevaluating and/or for obtaining exemptions from technical specifications. In that regard, this program will produce short-term interim products for use and will modify them, if required, to develop the final guidelines and implementation strategy for TS evaluation. In this section, the overall outputs at the completion of the program are defined.

The outputs to be generated are of varied nature and as such, the individual items listed in each category may overlap with items in other categories.

The major products being developed within the PETS program are the guidances for evaluating particular aspects of TS, their implementation program and the peer reviews. These are listed under the Categories A, B, and C below. However, in developing the guidelines and in demonstrating the guidelines through case studies, a number of related products will be developed which are essential for evaluation of TSs. These are listed under the categories D through G and include the recommended plant models, data base, TS evaluation models, measures and treatment of issues to be employed in TS investigations. In addition, future research needs to make TS analysis more robust will be defined.

A. Aspects of the TS That Will be Included as Products

Guidelines for NRC use include a list of procedures for setting AOTs and surveillance requirements, addressing requests for AOT extension, requests for extension of TIs, and procedures for determining under what conditions test after-failure requirements should be imposed. These guidelines will be structured using a flow logic scheme which will serve to provide a sequence of issues to be addressed, with alternate flow paths indicated by the different possible resolutions of each issue. Such a scheme is considered necessary because of the many possible interrelated issues that could affect the decisions to be made concerning these elements of the TSs.

The list of information and analyses to be supplied by a utility will be that required to resolve the utilities' request or the issue under consideration. This information may include plant models at the system or sequence level, documentation and assumptions relevant to the construction of these models, data base and assumptions concerning data relevant to the plant models, importance calculations, treatment of human error, common cause assumptions, and treatment of support system interactions and between-system interactions in the analysis.The utility will specify what codes were used and what risk measures were considered in their analysis. Complete documentation of assumptions and data included in their analyses should be provided. This list of information and analyses will be explicitly developed to a level which should eliminate the likelihood for misunderstanding. A trial application will aid in determining this level.

The guidelines for all of the aspects of TS treated in this program will be demonstrated by applying them to real systems and sequences, chosen from plant specific PRAs and the corresponding ASEP models. The ASEP models will be used for these demonstrations. In cases where the ASEP models may not be detailed enough, or deficient in other aspects, recommendations will be made for modifications to these models that will make them appropriate for TS evaluation. A data base will be developed for these demonstrations, and for future applications to TS. This data base will be a product from this program. The data base will include component failure rates, human error probabilities, suggested common cause data, test and maintenance outage times and frequencies, uncertainties, and bounds on all data, and recommendations concerning use of generic and plant-specific data. The demonstrations will include impacts on risks at various levels (e.g., system and sequence), the role and use of importance measures in evaluating and setting TSs, and the impact of data uncertainties and sensitivity analyses. The products resulting from the demonstrations will be AOT and surveillance requirements for the example plant systems. The demonstrations will serve both as the basis for synthesizing the guidelines, and to demonstrate and validate the use of these guidelines.

B. Peer Reviews as Products

Three types of peer reviews are generated as products from the TS program:

- Timely peer reviews of the Owners Group approaches to evaluating AOT and surveillance requirements,

- "Pinchpoint" common reviews of the TS program described herein by representatives of EPRI, the Owners Group, and the NRC, and

- A final, formal, peer review of the TS program described herein by representatives of interested parties, including utilities.

Each of these review types are serving to transmit information both from and to the TS program in a timely manner. However, they each serve somewhat different purposes.

The peer reviews of the Owners Group approaches serve to inform the staff of the PETS program, and the NRC, of the approaches being taken in the G.E. analysis. In addition, these reviews provide a forum for an exchange of views regarding the important issues to consider in implementing TS requirements, and should help in resolving any differences in viewpoints that might exist between the two groups addressing similar issues. These peer reviews are conducted on a schedule consistent with information being available from the Owners Group program.

The "pinchpoint" common reviews of this program by representatives of EPRI, the Owners Group, and the NRC allow each to provide input to, and influence the course of the PETS program. In addition, information and techniques generated by this program will be transmitted, in a timely fashion, to the interested parties. These reviews are currently ongoing with one meeting completed and a second scheduled for early November 1984.

A final, formal peer review of this program by interested parties, including the utilities, is recommended as a chance for final comments and information dissemination. This final peer review will result in finalization of the implementation plan. It will be conducted after completion of the PETS program.

C. Recommended Plant Models to be Employed in TS Investigation

Products include: an evaluation of and recommendations in the usefulness of the ASEP model and the modeling detail necessary for TS evaluation, any recommended modification to the ASEP models, and guidance on use of other plant specific PRAs for evaluating TSs when PRAs and ASEP models are not available for the particular plant under investigation. Recommended shutdown risk models will be provided, as well as guidance on using the shutdown risk models, and when shutdown risks should be considered to evaluate AOT and extensions of downtimes.

D. Data Base Products

Data bases appropriate for evaluating TSs are being defined. The data base definitions will include guidance in obtaining plant specific data, and guidance in using both generic and plant specific data for TS evaluation purposes. An interim data base will be assembled that includes component failure rates and models, uncertainty values, human factors and common cause data, and generic repair times for use when evaluating AOTs. The data base will be constructed from existing data bases, and from a limited review of plant specific information that is available at the present time.

E. TS Models as Products

Available computer models that may be of use for evaluating TSs are being examined and compared. The strengths and weaknesses of these models for evaluating TSs will be listed as a product of this program. It may be necessary to modify one or more of the existing models to make them more useable to evaluate TSs; these modified models will also be a product of the program. Recommended modes of operation and options will be a product of the program. In addition, it is necessary to have component repair models for evaluating TSs. Recommended component repair models will be a product of the program.

F. Recommended Measures as Products

There are many different measures and criteria of acceptance that could be used to evaluate and establish TSs. One of the products therefore will be to recommend those measures that are most useful for TS evaluation. The benefits and limitations of each of the measures for determining AOTs and TIs will be evaluated and provided as a product. The various levels at which AOTs and SRs could be evaluated (system, sequence, release category, etc.) will be addressed and recommendations made as to what level should be used. The measure(s) appropriate at this level will be recommended and provided as a product.

Measures appropriate for determining TS include cost/benefit, safety goals prioritization, importances, occupational risk, value impact, manufacturers warranties, shutdown risks, and reliability performance measures, to name a few. The role of each of these for evaluating AOTs and SRs will be addressed and specific acceptance criteria will be recommended.

G. Issues to be Resolved as Products

To complete the program, it will be necessary to address a wide variety of operational and reliability issues. A product will be recommended treatments for each of these issues for evaluating and establishing AOTs and SRs. Issues related to system reliability that will be addressed in the program, and for which recommendations will be included as products, are:

- The role of common equipment failure modes in establishing AOTs and SRs.
- Recommended treatment of human errors for TS evaluation.
- The role of equipment wear out, and treatment of wear out issues.
- The role of test-caused failures and test inefficiencies.
- The role of uncertainty and sensitivity analysis in TS evaluation.
- The role of recovery from test, maintenance, and equipment failures for assessing TSs.
- The role of failure causes and their impact on repair time estimates for AOTs and extension of outage times.
- Guidance as to when shutdown risk should be considered in the evaluation of AOTs and extension of outage times.

Issues related to plant operation that will be addressed and for which recommendations will be included as products, are:

- The role of various testing schemes and their interactions with the reliability related issues.
- The roles of reliability assurance reliability monitoring and their impact and benefits for TS evaluation.
- The roles to be played by inspection and enforcement.
- The role of multiple component outages and the role of TSs concerning this issue.
- The allocation of downtime as a viable TS method, and the impact of this scheme on risk.

4. SUMMARY OF CURRENT RESULTS

This section presents an overview of the documented products and direction of the PETS subtasks related to the program plan, AOT determination, and TI determination. Since the authors were requested by the NRC to present a top-level review of the program, this paper is not intended to delve into the proposed mathematical models or their development. In addition, it does not cover all pertinent points in the application of the interim approaches. Instead, it introduces the problem sets and the concepts of their solutions. If the readers desire additional detail, they are encouraged to review the three project reports listed below:

1. "PETS Program Plan," J. Boccio, et al., October 1984.

2. "Determination of Allowed Outage Times (AOTs) from a Risk and Reliability Standpoint," W. Vesely, July 13, 1984, Draft.

3. "Issues for Evaluating Test Intervals for Online Standby Safety Systems," E. Lofgren, September 20, 1984, Draft.

4. "PETS Program: Supporting Data Project Report," E. Collins and M. Jacobs, October 1984, Draft.

4.1 Program Plan

Task 1, the development of a detailed technical program plan has been completed and a final report issued, the results of which have undergone extensive peer review from the NRC staff and representatives of industry. In addition the plan has been presented to the IEEE subcommittee 3, Operations, Surveillance, and Testing at the request of its chairmen. The final report is currently available through the Division of Risk Analysis and Operations, NRC, or Brookhaven National Laboratory.

4.2 Determination of AOT's

Task 5 has developed an interim approach to the determination of AOTs and the review of AOT extensions, reference draft report, dated July 13, 1984. An AOT is the period of time during plant operation in which the component can be inoperable such as during scheduled or unscheduled maintenance. The AOT should be optimized to allow enough time to perform the needed operation on the component so as not to encourage a temporary fix, called "band aiding," but yet not too long as to increase the plant risk by component unavailability. During an AOT the probability of an accident can be increased due to the unavailability of an essential component. It is this potential increase that should be evaluated when reviewing specific case by case extensions to the predetermined AOT. The identified associated risks are:

- Operating Accident Risks
- Shutdown Accident Risks
- Shutdown Economic Risks
- Occupational Radiation Exposure Risks

The first two lend themselves to analysis by PRAs; the third should be investigated through nuclear plant outage models; the fourth potentially through plant actuarial records.

When looking at the question of regulatory approaches to the evaluation of AOTs, the NRC could specify the actual AOT value, specify the procedure, criteria, and plant data, or set only to the acceptance criteria. Each approach, placing a different burden on the NRC and on the licensee is discussed in detail in the draft report. In addition the pluses and minuses of each developed strategy and criteria level is presented to guide the analysis of a particular AOT. In general, the draft report utilizes time independent PRA techniques to quantify the risk of an AOT or change to an AOT and, where necessary, relies on time dependent tools such as the FRANTIC computer code for detailed specific analyses.

A decision process, Figure 4.1, has been developed which can be followed in deciding whether the operating accident risk can be focused upon or whether multiple risks need be considered. The decision process shown applies to an individual AOT being evaluated or to multiple AOTs. For the latter, the accident risk is the cumulative risk. Figure 4.1 does not explicitly show the case where the AOT has significant accident risk and also needs to be extended; however, this case can be simply deduced.

When an overall aggregate risk index is controlled, such as in a value impact or cost benefit analysis, then there is little option as to the level at which risks are examined and explicitly controlled. In general, the risks evaluated are the overall risks including the frequency of accidents and shutdowns and their consequences. More basic attributes can be focused upon, but these need to be translated to equivalent overall risks to have a common scale of comparison.

As discussed in the previous section, in many problems the single risk that needs to be examined to determine AOT acceptability is the operating accident risk associated with the AOT. The levels at which the operating accident risk can be focused upon and the particular risk characteristics which is focused upon are discussed in the July 13, 1984 draft report.

At the overall risk level, the accident frequency times consequence, or some other equivalent overall risk measure, is evaluated for acceptability. At the core melt or damage level, the core melt frequency is evaluated for acceptability. The level of evaluation progresses to more basic risk contributors until finally at the component level, the component unavailability would be evaluated for acceptability. The unavailability characteristics can include the failure to start and the failure to run. The particular measures which are used to quantify the risk characteristics are described in a proceeding section.

At a given level of control, criteria are established for the associated risk characteristic. Thus at the system level, criteria are defined as to what constitutes acceptable system unavailability associated with the AOT. At the accident sequence level, criteria are defined as to what constitutes an acceptable accident sequence frequency associated with the AOT.

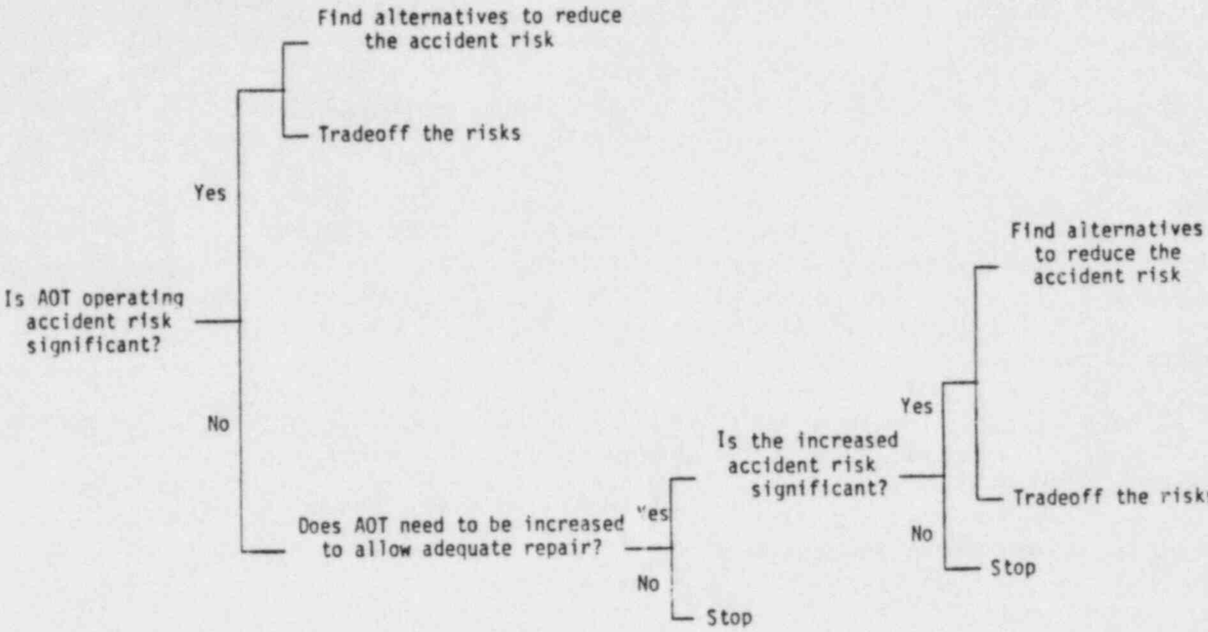

Figure 4.1 Decision process for evaluating AOTs.

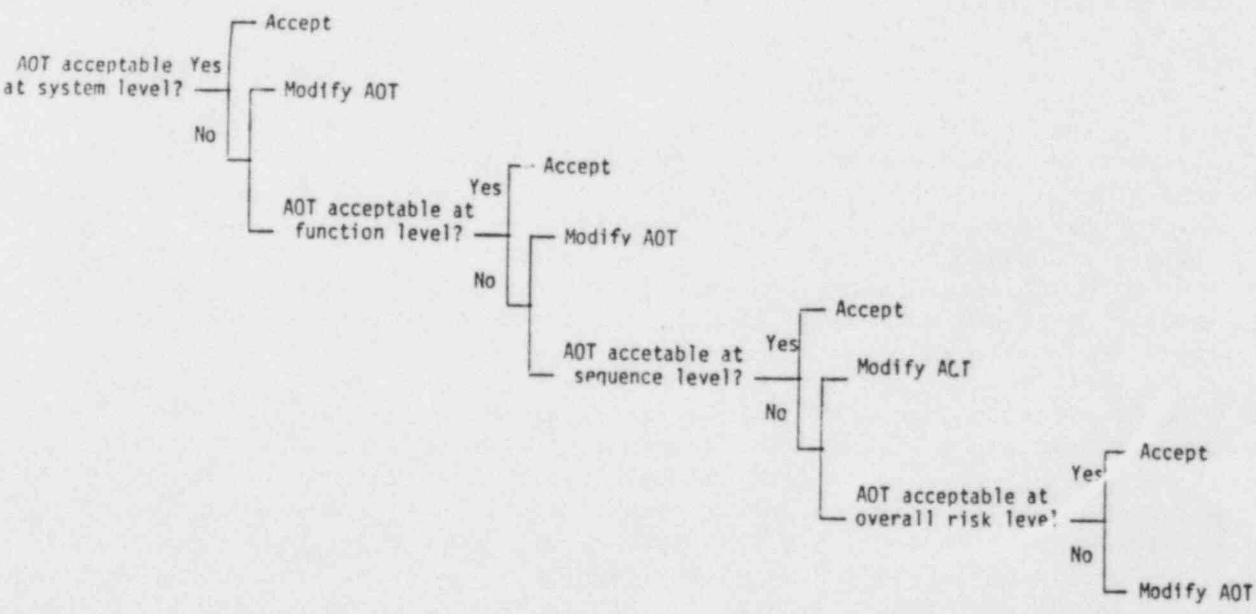

Figure 4.2 A stepwise level process for deciding if accident risk is significant.

The criteria can be flexible or can be quite prescriptive and can be absolute or relative in nature. In general, for relative criteria, the particular risk contribution associated with the AOT are compared to the accident risk associated with other contributions. If absolute criteria are used, the particular risk associated with the AOT are compared to some absolute criterion value.

In principle, the AOT operating accident risk can be controlled at any of the levels identified. Controlling the risk at one level controls all the risks at all higher levels excluding those system interaction contributions. The control may be initiated at any level and will result in control of the overall risk.

In practice, the level at which the accident operating risk is addressed and controlled depends upon implementation considerations. The higher the level at which risk is addressed, the more flexible will be the acceptable AOTs for components. For example, if criteria are established at an accident sequence level, then there would be various alternative acceptable AOTs for the components comprising the function.

Another advantage of controlling risk at a higher level is that the acceptability criteria will generally have to be less stringent. Bounding case situations will need to be considered at lower levels and logical relationships which may counterbalance the situation are not considered. For example, if a particular system enters into different accident sequences with different system success definitions, the most stringent definition will need to be used if criteria at a system level are established. However, the sequence with the most stringent requirements on the system may also contain other reliable systems which diminish the risk contribution of the system and the AOT.

The above are advantages of controlling risk at higher levels. There are also, of course, disadvantages. The higher the level at which risk is controlled, the less specific and less prescriptive will be the acceptable AOTs. This makes it more difficult to review from a regulatory standpoint. Also, as risk is addressed at higher levels, the models and evaluations generally become more complex with greater associated uncertainties. Attempting to compensate for these uncertainties may offset the advantages of going to higher level control.

In practice, the choice of the level at which risk is addressed involves a compromise among the various advantages and disadvantages. It is sufficient to evaluate risk at a low level if, even with conservatisms, an adequate AOT is provided which allows effective repair to be performed without causing significant risk. Allowing a larger AOT beyond that needed simply causes additional accident risk. If evaluating risk at a low level is not sufficient to allow an adequate AOT, then evaluating risk at a higher level may provide a larger AOT and still provide adequate risk protection.

If a fixed level of risk evaluation proves to be inadequate in application, an adaptive risk level may be used where risk is evaluated at only the level and complexity needed. Figure 4.2 shows this adaptive type of decision process starting from a system level evaluation; a component level evaluation could also be the start point. If criteria do not exist at a level, then that level would be skipped.

For a detailed presentation of the developed approach to determination of AOTs from a risk and reliability standpoint and the reliability model's development, the reader is referred to the July 13, 1984 draft report.

4.3 Determination of TIs

Task 6 is initially intended to take an eclectic view of strategies for specifying TIs. Options are identified and the advantages and disadvantages of each are discussed. It is not intended to specify a method for establishing TIs. Rather, the major issues involved with establishing TSs on TIs are identified and discussed with emphasis on how they impact the various strategies.

A request for extension of TIs may involve only one system. The issue is whether or not the extended test intervals would result in an unacceptable system unavailability. The system's importance to a particular function, sequence, or to core melt frequency would be an issue in this case. On the other hand, evaluation of test intervals for multiple systems impacts function, sequence, or core melt frequency directly. Acceptable test interval ranges could be obtained by staggered testing schemes among the systems, as well as by limiting the test intervals. Thus, it is necessary to simultaneously consider the impact of testing strategies and test intervals when evaluating TIs for multiple systems.

The major elements of the problem of evaluating and establishing TSs for TIs that are treated in this paper are the following:

- Acceptance Criteria
- Strategies for setting and evaluating test intervals
- Model and data requirements
- Proposed test interval evaluations.

A. Acceptance Criteria

As with the AOTs, there are seven possible levels at which test intervals could be evaluated: component, system, function, sequence, core melt, total risk, and total cost. To some extent, the level at which test intervals are evaluated will impact the contributions that can be included.

The type of safety goal to be used to evaluate test intervals could be either relative or absolute and could be based on either pointwise or average unavailability. Unlike the AOT evaluations, cumulative unavailability safety goals are not appropriate for evaluating test intervals (for the constant failure rate case), since the average and pointwise unavailability can always be estimated over one test interval. Thus, there is no concept similar to the AOT evaluations where one limits the accumulated risk from outages over some defined baseline period.

B. Strategies for Evaluating TIs

Strategies for evaluating TIs are based primarily on combinations of options from the factors that define the acceptance criteria. In addition to combinations of the acceptance criteria factor options, constraints such as manufacturers requirements for testing particular pieces of equipment, component wear-out from too frequent testing, and consideration of certain non-standard types of faults, such as human errors associated with testing that may not be detectable until a full-scale demand for the equipment occurs, are considered for developing test interval strategies. As previously stated, the viewpoint is taken that the strategies requiring less data and models, and small expenditure of effort to implement are preferred over strategies requiring large amounts of data, complex models, and therefore large expenditure of effort to implement. Such a viewpoint leads naturally to a layered approach to TSs.

C. Proposed Test Interval Evaluations

Based on the issues identified and discussed in the September 20 draft, a set of test interval evaluations using information from the Limerick PRA and the ASEP program are suggested. These evaluations are designed to test the sensitivity of acceptable test interval evaluations to the range of possible techniques and values upon which test intervals could be established. The applicability of the ASEP models and data for test interval evaluations will also be tested. The proposed evaluations cannot be considered to be complete in the sense that they represent all of the necessary evaluations for test intervals, since it is anticipated that additional evaluations will be suggested by the results obtained from the initial set. However, they provide a convenient initial set of analyses upon which to base further work. For further information on the current technical work in the area of TIs, reference the September 20, 1984 draft report.

5. SUMMARY

Nuclear plant Technical Specifications, issued as part of the plant operating license, contain both operational and surveillance requirements for assuring the safe operation of the plant. Current technical specifications have come under criticism as being, in some cases, inappropriate and even risk-increasing.

This paper briefly describes a program plan, prepared by BNL, to examine approaches, methods, and criteria for developing a quantitative basis for making engineering judgements in revising technical specifications.

A rather comprehensive program is described which, although stressing risk-based analytical development and techniques, takes cognizance of other engineering and operational aspects for assuring safe operation. As such a seven-task program is designed to interface with and impact on other NRC-sponsored and industry-sponsored programs. The end result will be a new methodology, along with a "procedures guide," for evaluating technical specifications and structured on a sound basis to expect only infrequent requests for extension.

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SOME THOUGHTS ON ALLOCATION OF RELIABILITY*

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ABSTRACT

The technical feasibility of allocating reliability to reactor systems, subsystems, components, and structures is discussed in this paper. The basic premise for this analysis is that a set of objective functions or safety variables has been defined on a global basis for a class of nuclear power plants. The decision variables, which represent the system, subsystem, component, and structural reliabilities are related to the global objective functions by a risk model obtained from an existing plant-specific probabilistic risk assessment (PRA). A multiobjective optimization technique is employed to obtain the set of decision variables which optimize (minimize) all of the objective functions. A cost function is introduced (and incorporated in the optimization scheme) which measures the cost of increasing reliability. Illustrative calculations were performed for a boiling water reactor with an existing PRA.

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One of the central themes of contemporary probabilistic risk assessment (PRA) is that the usefulness of PRA derives from the safety insights gained, the identification of plant design and operational vulnerabilities, and the potential role that PRAs may play as safety/risk management tools. It has also been widely said that the "bottom line" risk numbers are the least useful part of a PRA. The dilemma here is that in order to gain the insights, identify the vulnerabilities, and manage the safety/risk, one must go through the process of delineating and quantifying accident sequences and computing various risk indices, i.e., the very nature of the PRA discipline requires that the analyst proceed toward the bottom line.

This paper summarizes[1] the methods, results, and conclusions of an initial study the allocation of reliability in nuclear power plants. This work is an approach to probabilistic safety analysis which uses the concepts of bottom line risk indices and the results of the systems and operational evaluations developed in probabilistic risk assessments in a constructively interactive way. The product of this approach is the display of information related to cost and risks of a particular nuclear plant design as a function of the unavailabilities of its constituent components, systems, and structures. Additionally, such information can be displayed as a function of alternative design configurations and/or operational practices by using the methods described in Reference 1.

The study that is reported on here is an assessment of the technical feasibility of allocating reliability in a self-consistent manner to various levels of plant performance. Specifically, the analysis addresses the allocation of reliability to reactor systems, subsystems, components, and structures. Earlier work[2-7] that is pertinent to this study is discussed in Reference 1.

It is the conclusion of this study that allocation of reliability is technically feasible. The fundamental elements of the analysis that lead to this conclusion are threefold: 1) a global set of measures of plant performance (top level risk indices or "objective functions") which would be subject to a preference assessment by a decision maker; 2) a model or prescription for relating the global set of measures of plant performance to the specific set of measures of plant performance (system and component unavailabilities, etc. or "decision variables"); 3) a method for deriving a finite, manageable set of self-consistent relations between the global and specific sets of measures.

In this study the first element was identified to be the following global set: core damage frequency, expected acute (or early) fatalities, expected latent fatalities, and the cost of achieving a particular set of values for the first three members of the global set. There were several reasons for choosing the global set at this level of plant performance. First, this set is not plant-specific. Second, this global set is likely to be understandable by the policy-level decision makers. Third, this global set is commensurate with the level of safety criteria that have been promulgated by various parties who have an interest in nuclear power plant operation. We note, however, that our global set of measures are not regarded as prescribed safety criteria or safety goals. Rather, they are a set of attributes which can be studied, compared and traded-off by the decision makers.

Central to our approach is the identification and use of the fourth member of the global set, cost. It was recognized early in this study that the cost of achieving a particular set of values for the first three members of the global set represented a necessary dimension from the point of view of those who must make practical, real world decisions and from the point of view of those who must obtain feasible engineering solutions from the methodology presented in this study.

The second fundamental element, namely, a model which relates the global set to the specific set was identified to be the probabilistic risk assessments (PRAs) which derive top level risk values from plant-specific failures and vulnerabilities. The PRA model is the natural choice for this element because of the abundance of existing PRAs for various nuclear power plants, the level of detail contained in PRAs in the areas of interest to this study, and the potential for enhancing the insights already gained from PRAs by performing the type of study presented in this report.

The third element was identified to be a multiobjective optimization procedure[8] performed on the PRA model with the global set regarded as objective functions. The optimization approach was selected, in part, to reduce the multiplicity of possible solutions to the problem defined by the relation between the global and plant-specific set to a manageable handful and, in part, to obtain the best and most rationally acceptable subset from the multiplicity of solutions. Therefore, the concept of selection of noninferior solutions was introduced; with this concept, solutions which did not yield a relatively favorable value for at least one of the four members of the global set were rejected from further consideration.

The overall methodology has been demonstrated with a nontrivial model. While the model does not represent a complete, particular, realistic power plant situation, it does contain many of the essential features that would be required of an analysis of such a situation. Thus the analysis was conducted for the purpose of technical feasibility and therefore particular features were purposely retained or built into the model in order to test and examine the successes and limitations of the overall methodology. The model is based on a full scope PRA for a BWR/4 MARK-II nuclear power plant. The significant classes of accident initiators and sequences are represented in the model. Dominant cutsets are retained and system dependences are included. In addition, containment performance variables and a seismic accident sequence are studied.

The cost models for the various systems and components are idealized parametric functions, but which nevertheless exhibit the correct intuitive trends for such models. The scope of the project did not allow for the development of realistic component-specific cost functions. However, sensitivity studies were performed on the parametric and functional forms of the cost functions in order to gain familiarity with the implied trends for the global set.

The results of the model analysis are displayed in terms of the set of noninferior solutions[8] to the optimization problem. Thus, for each noninferior solution a set of global values (risk indices and cost) and a corresponding set of plant-specific values (system unavailabilities, etc.) are obtained and displayed. At this point the technical analysis of reliability allocation is complete. It would then be the choice of the decision maker to choose among the noninferior solutions by performing a value tradeoff or preference assessment.

The <u>full</u> display of information (as is illustrated in Reference 1) to the decision maker can be a guide in the selection process of those choices which warrant more or less consideration. For example, the full display can indicate the ranges of values for which particular global or plant-specific measures vary rapidly or slowly and thus the incremental value of alternatives can then be gauged by the decision makers.

Reference 1 provides a brief study of how to incorporate uncertainty into the allocation procedure and it concludes that a formal technique exists for a restricted set of uncertain parameters with specified distributional forms. Less formal approaches, based on sensitivity analyses and error propagation techniques, are also possible.

Some outstanding technical issues related to the allocation problems are:

1) How good are the existing PRA models for the purposes of the allocation procedure presented in this study? The concern is whether hidden dependences, lack of completeness, unrealistic assumptions, etc. would significantly mar the conclusions derived in the allocation procedure. Of course, PRA itself suffers to varying degrees from these shortcomings. Clearly, gross flaws in a PRA model would correspondingly limit the value of the results of an allocation procedure. However, it is not clear whether some of the particular assumptions, methods, and models to which PRA results are significantly sensitive, also lead to correspondingly significant variations in certain results in the allocation problem.

2) Can a useful cost model be formulated? Since the scope of this project precluded detailed investigations of how to formulate realistic cost models, the question must remain open. Further investigations would examine whether appropriate cost data can be gathered on system and component reliability improvement (and decrement) and whether the relevant costs are reflected in the cost models (cost completeness). Further investigation would examine the impact of discontinuous cost-reliability functions on the solutions of the allocation problem.

3) Are uncertainties adequately addressed? In PRA analysis itself, the adequacy of uncertainty analysis is a subject of active investigation. Reference 1 has attempted to address this problem in its discussion of

certainty equivalents[9] and in the approaches to uncertainty that are outlined in the report. Further investigation is needed in this area and these investigations should consider existing and new developments in PRAs.

4) Are there significant mathematical/computational problems with the current optimization techniques and does software exist or need to be developed to resolve these problems? The report[1] discusses a problem encountered with NOT GATES and with better than rare event approximations and their relation to global optimality of the solutions. This problem would need further investigation. In addition, if larger and more realistic models are to be analyzed, then the problem of computational speed may need to be addressed.

5) Are the results and insights obtained from the allocation problem plant-specific or generic? This question does not have a general answer. Clearly, because all U.S. reactors are unique machines, strict and wholesale generic conclusions are not valid. However, some trends and insights may apply to classes or groups of plants. Obvious differences in support system dependences, for example, would tend to preclude strict generic allocation of safety functions and of front-line systems. Nevertheless, some trends (i.e., ranges of allocation) may be discernable upon closer investigation. Of course, the value of this information will depend on its end use and therefore any further investigation ought to be pursued with this in mind.

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Practical Reliability Engineering Applications to Nuclear Safety

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Introduction

PRA studies have been successful in providing a quantitative perspective on the important contributions to risk and on the relative impact of potential hardware modifications and procedural changes in reducing public risk. They have also been successful in some applications in demonstrating that certain modifications or requirements can be deferred or eliminated with no significant safety impact and with a positive effect on cost or plant availability. This paper will consider the applications being made by utilities using PRA models and data, that will improve safety in operation and lead to a continuing demonstration that nuclear plants are achieving acceptably low risks. These applications are important to the industry because they achieve their objectives over a period of time and without expensive upfront modifications to the plant configuration. Moreover, they concentrate on the "how", rather than the "what", in contrast to the way in which PRA has been used to date, i.e., to focus on leading causes of risk.

The potential for the applications of PRA-study results, insights, and methods has been fully recognized; examples are the training of engineering, licensing, and operating staffs; evaluations of plant availability; evaluations of the risk of losing capital intensive equipment, and assessing the importance of operating incidents. While such applications have been made at least to a limited degree by several utilities, most PRA study sponsors have been preoccupied so far with completing their assessment of the public risk impact of backfits. Hence, benefits have not yet been fully realized.

It is clear, however, that both utilities and regulators will be moving towards realizing such benefits in the relatively near term. Since significant decisions will rest on conclusions from the various types of analysis involved, it is important that the boundaries of applicability of the analyses are clearly understood. The PRA Reference Document, NUREG-1050,[1], gives a careful account of the current understanding of these concerns and concludes that many applications of the probabilistic methods are relatively insensitive to their acknowledged shortcomings. To help minimize these uncertainties, a major emphasis in the EPRI R&D program[2] of the past three years has been to provide more systematic methodology in key areas such as common cause failures[3], human reliability[5], and seismic hazard[6] analysis.

These efforts to reduce and quantify uncertainties will continue although increasing emphasis is now being placed on the applications R&D that is the subject of this paper.

Systems Reliability Analysis and Technology Transfer

Table 1 shows some of the functions that can be served by PRA with an indication of their individual susceptibility to uncertainty.

The functions described as "Probabilistic Safety Design Review" and "Operating Safety Aid" are of particular usefulness to utilities, possess the largest number of applications, and are the least susceptible to uncertainty. The table shows that a Level I PRA will contain the basic information needed by a utility pursuing applications in these two areas. It suggests that systems analysis integrated with data acquisition and real time plant status information as depicted in Figure 1 can be of major assistance to plant management.

Systems Reliability Analysis is the backbone of the Level I PRA. Basically it represents the plant by a series of logic models that go beyond conventional systems analysis by 1) interfacing many system models, 2) providing accident sequence models (event trees), and 3) possibly including external event analysis. However, the particular applications noted in Table 1 may require systems analysis at a variety of levels, not all of which may be present in Level I PRA models. Modification of these models has to be considered for the particular job in hand, either to make them more sophisticated and detailed, or less, or to change their scope and objectives. For example, models of additional non safety-grade systems are likely to be needed for system interface studies, and Failure Modes and Effects Analysis (FMEA) for maintenance studies. Similarly, root cause failure analysis and special failure data recording and analysis efforts may be needed from time to time.

Clearly a Level I PRA will not answer all the needs. The development within utilities of a growing capability in systems reliability analysis will be absolutely essential if the promised benefits of such analysis are to be realized. In fact, most good quality PRAs have already discovered that close involvement of a number of utility operations and engineering personnel is essential to a quality product. The spearhead for the development of systems analysis capability in utilities to the present time has undoubtedly been derived from the quality assurance objectives of getting a PRA that accurately and realistically represents the current status of plant design and operations, and minimizes unnecessary conservatisms.

TABLE 1
POTENTIAL FUNCTIONS OF PRA

Measure Absolute Risk	Meausre of Relative Risk Contributors	Probabilistic Safety Design Review	Operating Safety Aid
1. Comparison between plants - risks (I)* - systems (II) 2. Identify risk outliers (II) 3. Compare with risks from other sources (II) 4. Use as licensing criterion (I)	5. Focuses on what counts and why: - what contributes to risk - what can reduce risk - what maintains low risk (II)	6. Quantitatively integrates system failure potential with safety scenarios (III) - identifies system configuration inadequacies - identifies system interdependencies - identifies man/machine interface inadequacies 7. Assists choice of remedy when inadequacy identified (III) 8. Independently checks conformance of design to license requirements (III)	9. Assists judgements and prioritization in operational strategy (III) - maintenance - tech specs - schedule for design changes 10. Assists operator training (III) 11. Provides framework for judgement of importance of operating incidents (III) 12. Provides technical basis for regulatory response (II)

* Difficulties in interpretation are introduced by lack of "scientific" validation and existence of large uncertainties. The importance of these for each application is indicated by a Roman numeral. The expressed uncertainties must be considered in relation to the limitations of not using PRA at all.

I Probably fatal
II Significant hindrance, careful sensitivity studies needed
III Not usually a significant problem

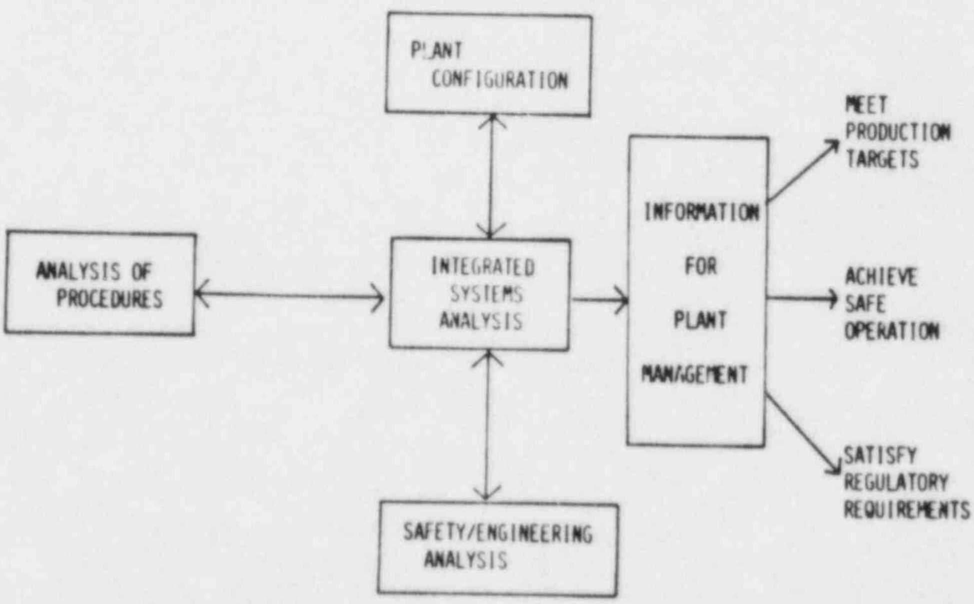


FIGURE 1
POTENTIAL SUPPORT FROM SYSTEMS ANALYSIS

In the future the utility capability in systems analysis will grow not only because of regulatory pressure to complete PRAs but also because the systems reliability discipline provides a quantitative basis for interaction with regulatory bodies, because of perceived applications to productivity improvements and because it is something that utilities can eventually learn to do largely by themselves. This last point is very important. The most effective and reliable systems analysis will be done by analysts who are close to, or themselves possess, experience in design, operations, production and the needs and objectives of their company. The most likely way this would occur in a utility would be through a cadre of competent engineering personnel, with good interfaces to Operations, Design and Production departments, who can apply the systems reliability technology.

It is certainly practicable to teach such engineers how to do fault tree or similar analysis and the modest statistical repertoire necessary for failure data analysis. However, it is a challenge to find personnel outside the utility companies who appreciate their day to day needs and priorities and who have up-to-date knowledge of the plants and the way they are being operated. The process of technology transfer is therefore central to realizing the full benefits of systems reliability analysis and at EPRI we have given it major emphasis and high visibility. We anticipate that a small group of suitably trained utility engineering personnel would be capable of dealing with the majority of the work that would arise for a nuclear plant. They would need only limited assistance from outside consultants who would deal with particular problems that would not yield to standard methodology.

At EPRI we have formulated our R&D program on this topic around such a supposition. Practical means of technology transfer at our disposal are 1) seminars and short courses, 2) code users groups and system modeling workshops and 3) R&D project cosponsorship. Each of these approaches is used extensively. For example, during 1983, 80 utility engineers from 30 companies attended fault tree, event tree, and GO reliability network modeling courses produced by EPRI. Both the WAM fault tree codes and the GO network codes have utility user groups (RPs 2507 and 818, respectively). The WAM users group comprises 26 members from 16 utilities and the GO group has 25 companies whose personnel are modeling systems in 20 plants. We intend to continue to actively maintain and support these codes and to continue the user group activities.

The culmination of this effort in the fault tree area is the WAME-02 package of fault tree analysis codes currently in a pre-release status at EPRI and undergoing validation and verification by the EPRI utility members of the WAME-02 Users Group. These codes will be available in 1985 as Production codes. The individual codes and their capabilities are summarized in Table 2.

TABLE 2

CODES INCLUDED IN THE EPRI WAME-02 PACKAGE

1. WAMBAM/WAMTAP:

   WAMBAM uses Boolean algebra minimization techniques to find the resultant logic expression from an input file and then calculates the associated point unavailability. It uses truth table methods to quantify the probability, which gives the code abilities to calculate more accurately than any other WAME-02 Series code. Furthermore, the WAMTAP option gives the code the ability to do sensitivity analysis.

2. WAMCUT:

   WAMCUT is used for the qualitative and quantitative evaluation of fault trees by obtaining and quantifying the minimal cutsets. It calculates first and second moments of the top event probability.

3. WAMFM:

   WAMFM is a post-processor program for WAMCUT-II to compute the point failure probability for the top event. WAMCUT-II performs a similar function as WAMCUT but uses a different algorithm for finding the minimal cutsets.

4. SPASM:

   SPASM has the capability of estimating the distribution, including the mean, of the system failure probability by Monte Carlo methods. The user inputs a system model and parameters.

Another software development project related to fault tree applications is designated as CAFTA - Computer Aided Fault Tree Analysis. CAFTA is software for a microcomputer interactive workstation which creates an environment to develop, debug, and update fault trees and companion failure data. Furthermore, CAFTA will interface with mainframe codes, such as WAME-02 codes, to quantify large fault trees and process the output from these large codes. In addition, CAFTA will facilitate the quality control of fault trees and fault tree data bases, a job that is essential but has proved difficult to do on large projects or on projects spread over a long time.

The CAFTA software should make fault tree analysis less costly, less time consuming, and of a higher quality. It will be made available to EPRI-member utilities in 1985.

The success-oriented GO methodology[7] is considered an emerging system reliability/availability assessment tool for the utility industry. The methodology employs straightforward inductive logic to model the functioning of engineering systems by using a collection of GO operators which simulate the behavior of various components found in nuclear power plant systems. The collection of GO operator types includes logical functions such as AND, OR, m-out-of-n gates and engineering functions such as switch, and normally open and normally closed valves. Figure 2 shows a GO model of a simple system displaying the logic operators directly substituting for hardware. The system success criterion is modeled directly at the system output.

Following are two examples of GO User Group applications.

Sequoyah Demonstration Study

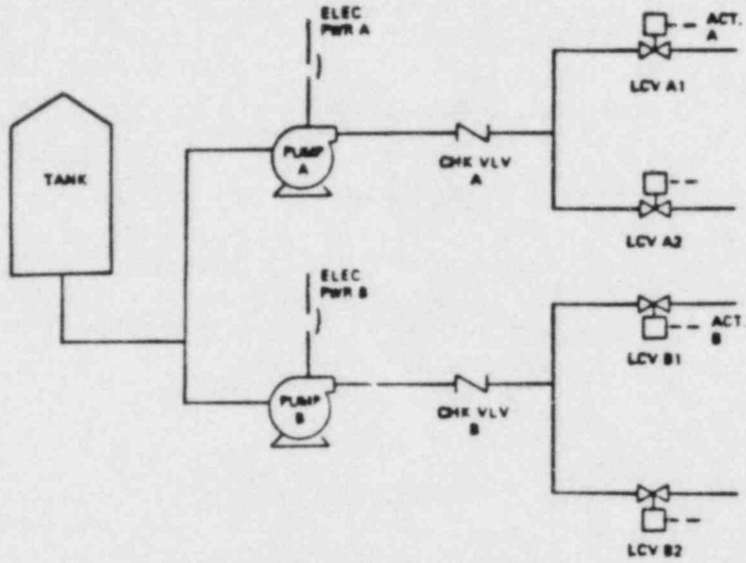
The Sequoyah full-scale nuclear power plant availability and safety study[8] has been a jointly sponsored effort between TVA and EPRI (RP1842). The study has conclusively demonstrated that GO methodology can be effectively used in a large-scale application, as an alternative probabilistic system analysis technique to the fault tree approach for determining plant availability and core damage frequency.

The availability analysis has been focused on operation at 100% rated power plant production. Any failure which causes a reduction in power output from full production has been included in the analysis. The overall plant availability model consists of three major parts: primary systems, secondary systems, and auxiliary systems. An integrated plant-level GO availability model, which is composed of over twenty-five production systems, has been developed and quantified; the major plant unavailability contributors have been identified.

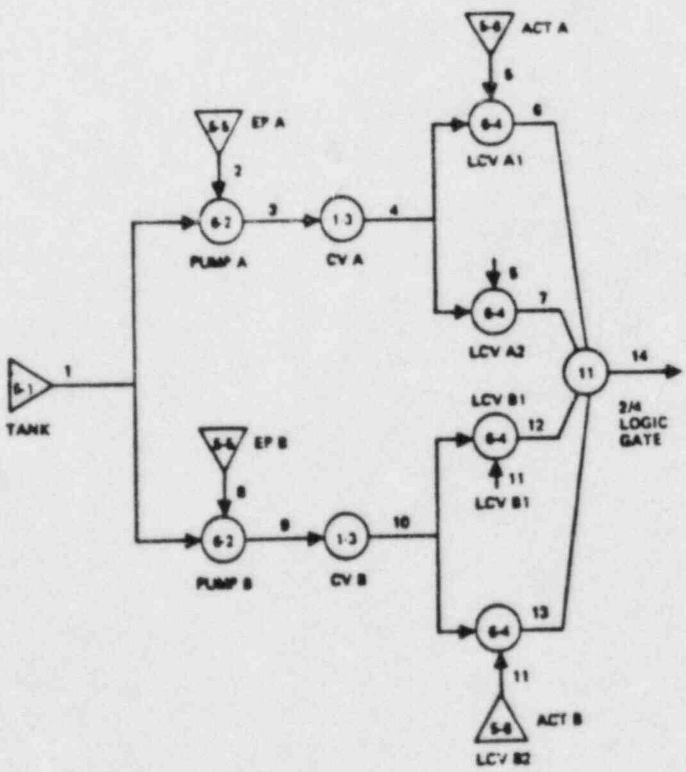
In the safety analysis, a total of fourteen initiating events have been examined. The GO representation of Event Sequence Diagrams (ESD) of each initiating event has been developed. The relevant GO systems models, such as safety systems and auxiliary systems, have been integrated into the GO ESD model for quantification, identification of critical component and dominant minimum cut sets and development of conventional event tree representation of the GO modeling results. External events, common cause, containment systems, and most of the operator recovery actions in accident sequences, have been excluded from the demonstration analysis. The study is approaching completion; the final report should become available soon.

FIGURE 2: GO MODEL COMPARED WITH SYSTEM DRAWING

EXAMPLE SYSTEM



GO CHART



Standardized Modular GO Subsystems Models

A variety of plant types and system designs are presently available in the electric utility industry. While the configuration of different plants and their systems is unique, there are frequently similarities among the systems in terms of subsystem configuration and arrangement of components. A number of recurring subsystems configurations have been identified in various GO system analyses. GO models have been developed to characterize these recurring system configurations. These subsystem models are termed standardized GO subsystem modules. It has been demonstrated in several EPRI and utility cosponsored studies (RP818) that these modular models can significantly facilitate the GO users in developing and analyzing system level models.

Currently, eighteen of EPRI's member utilities have utilized GO methods using their own personnel to study various problems. The models developed during this period, for example, on main and auxiliary feedwater, high pressure injection, core spray, component cooling water and AC power systems, were used for reliability evaluations of as-built plant systems, methods validations of earlier safety analyses, and evaluations to support positions on licensing issues.

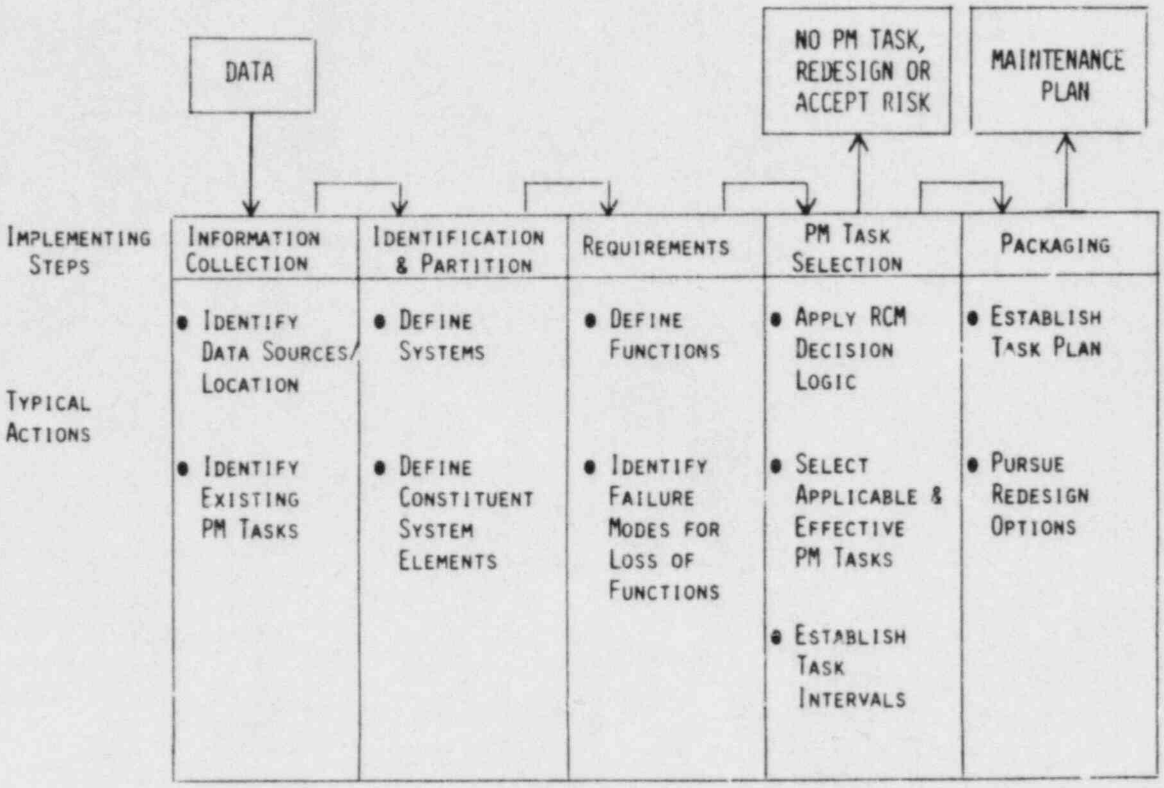
We believe the popularity of the GO method among utilities for ongoing system studies can be attributed to some very basic characteristics that fault trees do not possess. They are 1) appearance of hardware in the models, connected in more or less the same way as in the system drawings, 2) the models can easily be modified to reflect configuration changes, and 3) the modeling capability is extremely flexible.

Reliability Centered Maintenance

Recently EPRI surveyed the commercial aviation industry to determine whether any of its successful practices might be transferred in a cost-effective fashion to the commercial nuclear power industry[9]. One of the results of this work was a decision to undertake a pilot study to apply Reliability Centered Maintenance (RCM)[10], a technique described earlier in the conference by Thomas D. Matteson, to a nuclear power plant system. A simplified schematic of the tasks involved is shown in Figure 3.

FIGURE 3

RELIABILITY CENTERED MAINTENANCE PROGRAM IMPLEMENTATION



Florida Power and Light Company's (FP&LC) Turkey Point Units 3 and 4 were selected as the plants to be studied, and the project (RP2508-2) was begun earlier this year. Initially the instrument air system was studied. However, because of the relatively low preventive and corrective maintenance load on this system, another more representative system was later chosen. A functional failure analysis of the component cooling water system has been completed, and the RCM logic tree, Figure 4, is currently being used to define potential preventive maintenance tasks that are both applicable and effective.

The FP&LC computerized maintenance data system (GEMS) has proven to be a significant aid to realism in the functional failure analysis. The initial experience with the instrument air system has also suggested methods for screening systems so that the most productive allocation of engineering effort is made.

A report will be published in early 1985 describing the experience with this pilot application and the potential preventive maintenance modifications. A second application to gain further experience with this technique is planned for 1985.

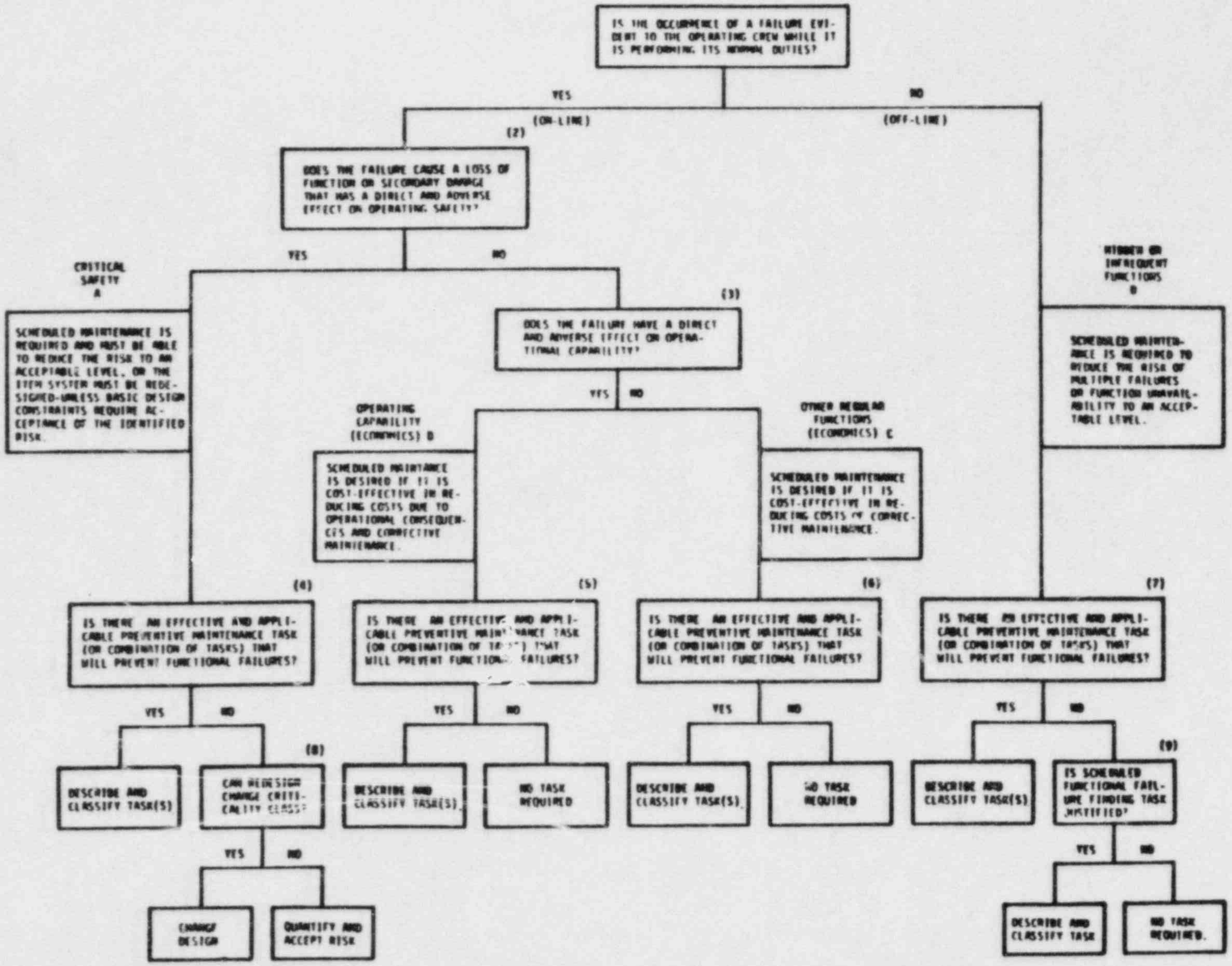


FIGURE 4

The RCM Decision Logic Tree

Analysis of Technical Specifications

Part of the effort by EPRI to apply probabilistic risk assessment methods and results to the solution of utility problems involved the investigation of methods for risk-based analysis of technical specifications. The culmination of this investigation (RP2142) is the SOCRATES computer code developed to assist in the evaluation of technical specifications of nuclear power plants. The program is designed to use information found in PRAs to re-evaluate risk for changes in component allowed outage times (AOTs) and surveillance test intervals (STIs).

All U.S. nuclear plants must operate in compliance with technical specifications, which are restrictions or requirements deemed necessary to protect public health and safety. "Tech Specs" include, 1) limiting conditions for operation (LCO) including allowed outage time for equipment, 2) surveillance testing requirements, 3) safety system set point limits, and 4) administrative controls. Only tech specs in groups 1) and 2) are being addressed in this project.

No uniform technical basis for establishing tech spec requirements exists. However, there is increasing interest on the part of industry and regulators to establish the risk-significance of tech spec requirements, thereby verifying that they achieved their intended purpose. Evidence indicates that such evaluations will likely show that certain requirements can be relaxed with no significant adverse safety impact but with significant positive benefits including cost savings and improvements in availability and operational flexibility.

The SOCRATES code is designed to process, with a minimum of re-analysis, cutsets which might already exist from a PRA or from a system or functional reliability analysis. Depending on the nature of the cutsets, SOCRATES can address 1) safety system or function unavailability, 2) core damage frequency, or 3) significant radionuclide release frequency. The code is specifically designed for tech spec analysis; and thus has advantages, capabilities, and flexibilities beyond other methods of assessing the safety significance of tech spec changes. Table 3 gives a summary of these advantages.

TABLE 3

Differences From Existing Computer Programs

0 Allows more detailed analysis of allowed outage time and test interval contribution to risk,

0 Evaluates allowed outage times and test intervals simultaneously for any group of components,

0 Allows evaluation of interactions between allowed outage times and test intervals,

0 Uses explicit equations, fast, and user-friendly,

0 Allows different testing schemes, and

0 Evaluates conditional, unconditional, maximum and cumulative risks.

The program has a very detailed component unavailability model which allows evaluation of three individual segments of a component's unavailability cycle as shown in Figure 5:

(1) Test phase - a scheduled downtime period for testing or maintenance,

(2) Downtime phase - an unscheduled downtime period for repair or maintenance, and

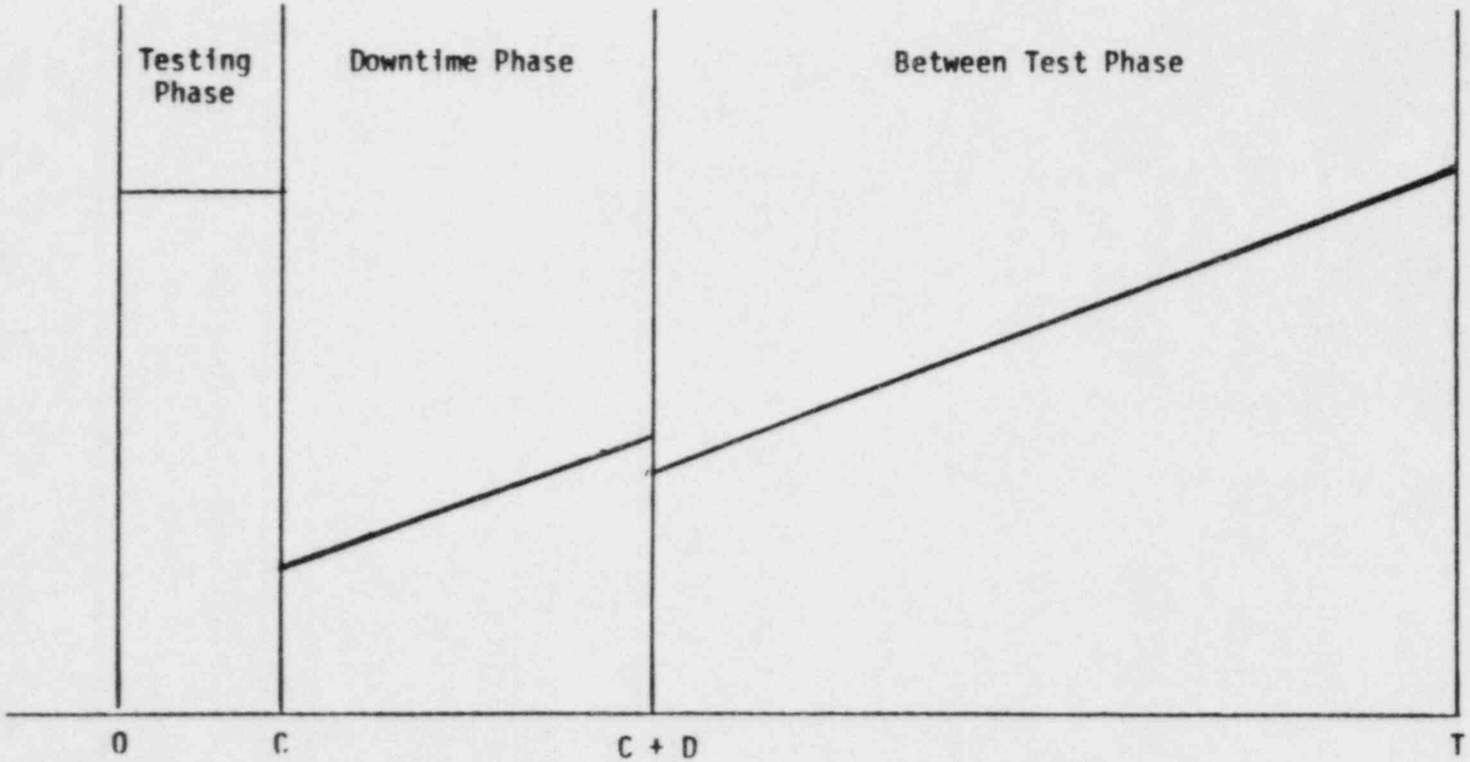
(3) Between test phase - the period between scheduled tests and maintenance in which failures are not detected.

Table 4 gives details of the component unavailability models in each phase. In addition to a constant probability of failure on demand and an exponential per hour failure rate, the component model includes terms for test-caused failures, failure to detect and repair faults, and a K-factor that allows the user to modify the rate that the component enters the allowed outage time. This allows the user to model the case where the component is taken into the AOT due to degraded performance in addition to component failures.

While the three phase model discussed above readily describes periodically tested components, the model easily adapts to include:

(1) Monitored components,

(2) Non-tested (non-repairable) components,



$$q(t) = q_0 + (1 - q_0)\rho + (1 - q_0)\,Q \qquad Q' + \gamma + \rho + \lambda(t - c) \qquad Qr + \rho + \gamma f + \lambda(t - c)$$

FIGURE 5. COMPONENT UNAVAILABILITY MODEL

TABLE 4 COMPONENT UNAVAILABILITY EQUATION

$$q(t) = q_0 + (1-q_0)\,\rho + (1-q_0)Q \qquad 0 \le t < c$$

$$q(t) = \rho + \gamma + Q' + \lambda(t-c) \qquad c \le t < c+d$$

$$q(t) = \rho + \gamma f + Qr + \lambda(t-c) \qquad c+d \le t \le T$$

$$Q = \frac{\rho + \gamma f + \lambda(T-c)}{(1-r)} \qquad Q' = \frac{\rho + \gamma f + k\lambda(T-c)}{(1-r)}$$

q_0 = probability that the test cannot be overridden on demand

Q = probability the component enters the test failed

γ = probability of a test-caused failure

λ = component failure rate

$k\lambda$ = rate that the component enters the allowed outage time

r = the fraction of Q that is not detected during the test and not repaired before the next test

f = the fraction of γ that is not detected and not repaired before the next test

ρ = probability the component fails on demand

c = scheduled allowed downtime (testing time)

d = allowed outage time

T = the total test interval, i.e., the time from the beginning of a test to the beginning of the next test

(3) Constant per demand components, and

(4) Components having a constant per demand contribution plus an exponential per hour failure rate.

The model can employ a realistic repair model to calculate the estimated unavailability based on utility data and experience. Alternatively, the user can conservatively assume that the entire AOT will be used to restore the equipment. These flexible features allow the program to accommodate most component types found in practice.

A unique feature of SOCRATES is the extensive use of specific testing schemes in calculating the time-averaged unavailability of minimal cutsets. While most components in PRAs are treated as independently tested components, SOCRATES handles:

(1) Simultaneous testing where the component test phase, downtime phase, and between test phase coincide,

(2) Sequential testing where the component test phase is offset so that the components are tested immediately after one another, and

(3) Staggered testing where the component test phases are offset such that the tests are evenly placed throughout the test interval.

Currently, SOCRATES will handle up to four components in a single test scheme while allowing arbitrary combinations of testing schemes within a minimal cutset. Different testing schemes within a minimal cutset are modeled as being independent of one another.

The input data requirements for the program are in three categories: cutsets, component unavailability parameters, and tech spec strategies. SOCRATES combines the component and testing scheme models to calculate numerous outputs that fall into four general categories:

(1) Unscheduled downtime risk measures - these measures are calculated from all contributions that have at least one component in the downtime phase.

(2) Scheduled downtime measures - these measures are calculated from all contributions that have at least one component in the test phase, but no components in the downtime phase,

(3) Between test measures - these measures are calculated from all contributions that have no components in the downtime and test phase, and

(4) Total measures - these measures include all the above contributions.

All four categories include measures of unconditional risk, i.e., time-averaged unavailability or accident frequency, depending on the minimal cutsets that are used. The risk measures can be obtained in absolute, differential, and relative form.

The unscheduled and scheduled downtime categories also include measures of conditional and cumulative risk. The conditional risk measure assumes that at least one component is in the downtime (or test) phase; that is, the component is assumed down with an unavailability of one. The conditional risk is the risk experienced for the duration of the component downtime. The cumulative risk is the integral of the conditional risk over a specified time period, accounting for the frequency of occurrence of the component downtime. These measures are available in differential and relative form.

The program also calculates an upper bound maximum risk. The maximum risk is accompanied in the output by the cutsets that contribute more than 5% to the total maximum risk and by the times in the cutset test cycles where their maximum occurs. These outputs are summarized in Table 5.

The program permits the user to calculate all of these measures as a function of downtime, test interval, or any parameter in the model for any group of components defined by the user. When calculating conditional quantities, up to five components can be assumed down in any single analysis. This allows a comprehensive evaluation of risk as a function of downtime or test interval with complete analysis of sensitivity to changes in failure rate, probability of failure on demand, or any component parameter.

The SOCRATES program is a unique and important tool for technical specification evaluations. The detailed component unavailability model allows a detailed analysis of AOT and STI contributions to risk. Explicit equations allow fast and inexpensive calculations. Because the code is designed to accept ranges of parameters and to save results of calculations that do not change during the analysis, sensitivity studies are efficiently performed and results are clearly displayed. Finally, the program is expressly designed for technical specification analysis by accommodating realistic testing schemes and component dependencies associated with the tech spec conditions. Furthermore, output options are tailored for this application.

Two applications of SOCRATES are being performed to exercise the methodology and test the code. The general philosophy adopted in such applications is shown in Table 6 and assumes the utility will propose alternative strategies which need to be evaluated.

TABLE 5

COMPUTER PROGRAM OUTPUTS

- AVERAGE VALUES

  R_D = ALLOWED OUTAGE TIME CONTRIBUTION

  R_C = SCHEDULED TESTING TIME CONTRIBUTION

  R_T = BETWEEN TEST CONTRIBUTION

  R_{TOT} = TOTAL CONTRIBUTION

- RISK CHANGE FROM REFERENCE VALUE (RV)

  $R_D - R_{D,RV}$ = CHANGE IN ALLOWED OUTAGE TIME CONTRIBUTION

  $R_C - R_{C,RV}$ = CHANGE IN SCHEDULED TEST TIME CONTRIBUTION

  $R_T - R_{T,RV}$ = CHANGE IN BETWEEN TEST CONTRIBUTION

  $R_{TOT} - R_{TOT,RV}$ = CHANGE IN TOTAL CONTRIBUTION

- RISK INCREASE

  $(R^+ - R_0)D$ = RISK INCREASE FROM OCCURRENCE OF ALLOWED OUTAGE TIME

  $(R' - R_0)C$ = RISK INCREASE FROM OCCURRENCE OF SCHEDULED TEST TIME

- MAXIMUM RISK

TABLE 6

RISK-BASED EVALUATION OF TECHNICAL SPECIFICATIONS (EPRI RP2142-1)

Proposed evaluation process for applications

1. Select "case studies" which represent real industry concerns,
2. Utility provides appropriate model solutions (cutsets) for the evaluation,
3. Utility provides failure data to quantify the model solutions,
4. Tech spec change strategies proposed with the utility,
5. Strategies evaluated by the tech spec analysis code, and
6. Results evaluated from several perspectives.

Southern Electric International and Battelle are evaluating a change in the allowed outage times for diesel generators at the Hatch Unit 1 plant of Georgia Power. Cutsets represent the loss of offsite power core damage sequences. SOCRATES is used to evaluate three alternative strategies for changing the allowed outage times for the diesel generators. Strategy 1 is a simple extension of the current LCO of 72 hours to some longer time. Strategy 2 is elimination of the existing LCO restriction and its replacement by an annual unavailability cap of 45 days per diesel. Strategy 3 is a modification of the existing LCO restriction to allow one 18 day outage annually while otherwise retaining the 72 hour LCO.

Duke Power Company and Battelle are evaluating changes in allowed outage times and surveillance test intervals for their high pressure injection (HPI) system at Oconee Unit 3. Core damage cutsets from the completed Oconee PRA[11] will be used for the analysis. Alternative tech spec strategies include extension of the existing 72 hour LCO on pump trains and of the 90 day test interval. This analysis will also address tech specs on the Engineered Safeguards system which actuates the HPI system. This study is focusing on the core damage frequency with one HPI unavailable. Both applications are testing many of the SOCRATES features and are attempting to use actual plant data and tech spec requirements in their analyses. Preliminary results of the Hatch study will be available in November. Results of the Duke work will be available in December.

Reliability Analysis Program With Integral Data, RAPID

Recently, some utilities have begun development of GO and fault tree models to evaluate the impact of component failure upon plant safety and system availability. Commensurate with these efforts, EPRI has also initiated a project with broad objectives (RP2508) entitled, "Use of PRA Methodology for Enhancing Operational Safety and Reliability". This project is a first attempt to apply system analysis techniques in a real time operational environment.

A basic premise of this new effort is that current generation nuclear power plants have increasing access to a significant body of useful information which, if effectively utilized, can enhance operational safety, maintenance, reliability, and coherent usage of the plant-specific data base for various engineering activities related to safety and reliability. This project is intended to develop and demonstrate integrated system software which will enable plant operational personnel and engineering staff to access appropriate data bases, system models, computer codes, operational requirements, and procedures for assistance in various decision making processes.

The study consists of two major activities: 1) to develop user-friendly, integrated computer software acting in some sense as an Expert System, and 2) to demonstrate the application of this software on-site. This integrated software, Reliability Analysis Program with Integral Data, (RAPID), is envisaged to consist of three interrelated modules as shown in Figure 6: 1) an executive controller which will provide engineering and operations staff with interface and control of the other two software elements, 2) a Data Base Manager which can acquire, store, select, and transfer data, and 3) Applications Modules which will perform the specific reliability-oriented functions. A broad range of these functions has been envisaged, for instance:

- Real-time, query mode, equipment/technical specifications compliance, and evaluation,
- Real-time, query mode, critical equipment and system status monitoring,
- Reliability and root cause data compilations (LER and NPRDS),
- Off-line, risk-based or reliability-based engineering analysis,
- Reliability-centered maintenance support, and
- Off-line, LCO and surveillance testing intervals evaluation and optimization.

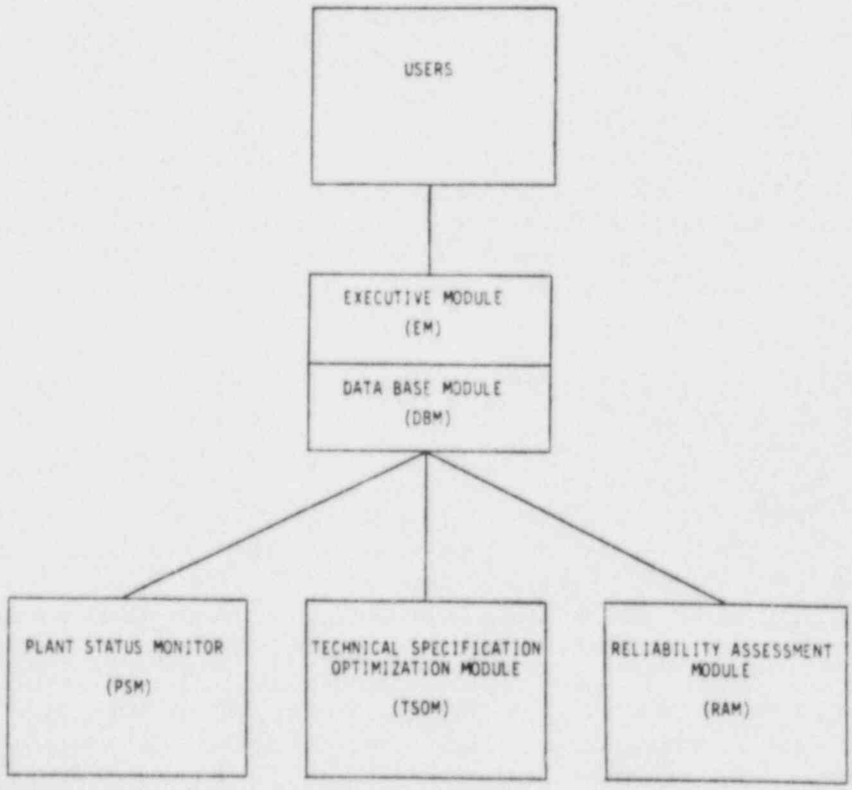


Figure 6 RAPID System Overview

The immediate emphasis will be focused on three particular application modules: a Plant Status Module, a Technical Specification Optimization Module, and a Reliability Assessment Module. The Plant Status Module (PSM) is intended to aid Operators with decisions regarding the acceptability and suitability of performing maintenance. The PSM has the following functions:

0 Display "up-to-date" equipment status information obtained from tagout administrative controls, and use it to derive the operational state of the plant, plant functions, systems and trains. The hierarchical PSM structure is plant, followed by plant functions, followed by systems, followed by trains, and followed by components,

0 Analyze limiting conditions of operation (LCO) and conflicts by identifying LCO compliance status and identifying conflicting LCO requirements,

0 Display surveillance requirements and their fulfillment, and

0 Provide configuration input for system availability and reliability impact studies.

The Technical Specification Optimization Module (TSOM) provides the capability for evaluating results of systems and sequence analysis to determine the sensitivity of the models to technical specifications and LCO requirements. This will enable users to investigate technical specification requirements with the objective of optimizing them with respect to system and sequence unavailability. In particular, the TSOM shall manage input and output data for technical specification optimization codes such as SOCRATES and FRANTIC.

The purpose of the Reliability Assessment Module (RAM) is to assist plant personnel in assessing the reliability of the plant and its systems. RAM will utilize detailed system reliability models and data to: 1) assess the impact of hardware or operational changes on system reliability and plant risk, and therefore to provide an aid to decision making regarding such modifications, and 2) help assess the impact of potential operational actions on system reliability e.g., taking a component out of service for maintenance.

The three applications modules described above will be integrated into the RAPID system with the Data Base and Executive Modules as shown conceptually in Figure 6. The Executive Module will provide the user interface for the system and translate user commands into instructions for executing the remaining modules. Users will interact with the system through a hierarchy of menus. The Data Base Module will provide management and storage of data, models and codes used by the three applications modules.

While the three applications programs will be developed for use by both plant operational staff and engineering/licensing staff, emphasis will be placed on making the PSM readily usable by the plant operational staff. Thus, additional requirements are being applied to the PSM to provide it with a simple, rapid and reliable interactive capability. Although the system is targeted for main frame computer operation, other hardware configurations will also be accommodated.

The programmatic structure of the project includes proof-of-principle studies at member utility plants, application demonstration, on site, of the software modules as they are developed and ultimate technology transfer and commercialization plans. A utility steering group has been formed to provide guidance on user needs, priorities and hardware compatibility.

Conclusion

The subject of plant availability improvement and its relation to safety is very complex. Undoubtedly, systems analysis techniques can help focus on and plan in-plant actions that lead to increased operational flexibility and cost effectiveness as can be seen from the objectives of the RAPID and SOCRATES programs. A better basis for decision making that improves both equipment up-time and the operators' awareness of risk sensitivities in operating plants, can be expected at least to reduce downtime due to tech specs, violations of tech specs and reactor scrams arising from inappropriate maintenance intervention. In general, this will lead to higher capacity factors as well as satisfying safety objectives.

More direct attacks on the unavailability of process systems might also yield safety improvements although this area is somewhat speculative. For example, enhancements in main feedwater reliability would reduce the demand burden on auxiliary feedwater systems. This would result in overall improvements in feedwater reliability and a reduction in challenges to many other safety systems. The law of diminishing returns would intuitively suggest that this is a more cost effective way of improving feedwater reliability from a safety perspective than trying to improve already reliable auxiliary feedwater systems. Fundamentally, the direction indicated here is to use systems reliability analysis to provide a more equal balance in reliability between safety and non-safety systems. Apart from what can be done to improve availability in existing plants, it would seem this subject is of major interest for new designs.

This paper has concentrated on applications of systems analysis by utilities. Underlying the potential benefits already described is the expectation that regulators will also use the technology to justify a relaxation in requirements that are too strict and the elimination of requirements that are irrelevant or counter to safety. Whether such modifications are made independently by the regulators or in response to utility requests, the quantitative nature of systems reliability analysis should enable rational decisions to be made.

In conclusion, system reliability analysis within a general PRA framework can probably be very useful in showing plant personnel where to put their resources and how to prioritize them to gain maximum effectiveness in maintaining low risk over the long term. To gain acceptance in the industry, in this mode of use, each application will need to have clearly defined objectives, address specific problem areas and stand on its own merits. In addition to improving safety and plant productivity, such applications will have a good chance of bringing relief from regulations that are more stringent than necessary.

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OPERATIONAL SAFETY RELIABILITY

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Introduction and Overview

Although many reliability efforts were underway by 1979, the TMI incident in March of that year and the investigations which occurred in its aftermath prompted a plan by NRC [1], as well as studies by DOE and EPRI [2-4] to investigate Reliability Programs as a cost effective way of maintaining LWR safety. The studies focused on revealing how the aerospace, military, commercial aircraft, and commercial air carrier industries approached optimizing safety, reliability, and costs of key systems. Additional motivation for more extensive NRC investigations was supplied by the insights gained from the Indian Point hearings, the Salem incident reviews, and the ATWS rulemaking [5-7] process. These efforts resulted in strong recommendations for the establishment of Reliability Programs with varying descriptions. Recent NRC research has identified Reliability Program elements practiced in other industries with having potential for use in the nuclear industry [8,9]. Studies to date have made only rather broad generalizations on the benefits or costs of such programs or their activities. No study has developed and then actually integrated a Reliability Program with existing utility operations programs for potential industry-wide implementation.

This paper provides a brief summary of research to develop an Operational Safety Reliability Program structure and selected activities for potential integration into the operation of light water reactors. The purpose of the program is to provide an alternative means of responding to the safety question "How can an acceptable level of plant safety be maintained over the lifetime of the plant?". First, this paper traces a background for Reliability Program development by summarizing the recommendations of other groups formed to address various safety issues. It then summarizes our findings on operational safety issues amenable to address by various features of a structured Reliability Program. Finally, it relates our conclusions to date on a

Reliability Program structure and describes future research to develop the activities that cost-effectively meet the purpose of the program.

Background

As cited above, the TMI accident provided the impetus for Reliability Program investigation. The majority of recommendations from the TMI-2 Lessons Learned Task Force Report [10] published in October, 1979 directly relate to the reliability of safety important systems and operational safety with a principal conclusion being that

> ..."although the accident at Three Mile Island stemmed from many sources, the most important lessons learned fall in a general area we have chosen to call operational safety. This general area includes the topics of human factors engineering, qualification and training of operations personnel; integration of the human element in the design, operation, and regulation of system safety; and quality assurance of operations".

The TMI Action Plan [1] of August 1980 discussed the development and use of reliability engineering programs:

> "Reliability engineering techniques can complement quality assurance and provide a disciplined approach to multidisciplinary systems engineering in the design of nuclear power plants, the development of startup test procedures, and development of operating, maintenance, and emergency procedures, and in operations. Criteria and procedures will be developed by NRR to apply reliability engineering practices to nuclear plant activities on a comprehensive and consistent basis."

The regulatory objective for reliability programs was identified in the NRC's Long Range Research Plan of March 1981 [11]:

> "The basic objective of systems and reliability research is to bring nuclear regulation into better congruence with the risks; that is, to identify and close gaps in regulatory requirements in risk-significant areas, to identify instances of off-target or unnecessary regulations, and ultimately to stabilize the licensing process."

A Reliability Program has also been proposed as a partial solution to the ATWS situation. An NRC Policy Statement (7590-01) dated November 1981 stressed the need for incorporating risk analysis information and related criteria for a unified safety and reliability program that could be broken down into risk-analysis and reliability-related categories. More recently, in April 1983, an NRC Task Force reviewing the ATWS situation at the Salem reactor issued a report [6] which identified several reliability deficiencies. It concluded that more attention must be focused on the reliability aspects of the reactor trip system (RTS) at all plants. The most recent guidance is in the Statement of Considerations for the ATWS Rule [7] in which, instead of requiring a Reliability Program for the reactor trip system as recommended by the Salem Task Force, NRC urges its voluntary development by licensees.

Guidance on the integration of PRA and Reliability Program elements is provided in the Indian Point Licensing Board Recommendations [5]. Unavailability evaluations for key systems are recommended to be routinely upgraded and utilized in day-to-day plant operations as well as personnel training.

One list of reliability techniques used in the military, aerospace, and commercial aircraft industries, as well as in the nuclear industry, was recently compiled in NRR research. The techniques were arranged according to the broader categories of management, design assurance, component availability, reliability degradation control, operational reliability, and experience feedback. Similar lists could be derived from the aforementioned DOE and EPRI reports [2-4].

Repeating themes in the aforementioned recommendations for a Reliability Program approach to operational safety included the following:

-- Adapt techniques and practices proven in other high technology programs
-- Use quantitative risk-based performance standards
-- Perform risk/reliability analysis of risk-dominant safety systems
-- Concentrate on dependent failures and root causes
-- Monitor systems to assure non-degradation of inplant performance
-- Monitor industry to broaden awareness of potential problems
-- Integrate risk/reliability insights into day-to-day operations.

This last item serves to factor proper operational safety reliability considerations into the requirements in Technical Specifications, test and maintenance strategies, system configuration, and actual performance of operating and maintenance functions. These themes will be shown to be reflected in our research to date and our preliminary conclusions on Reliability Program structure.

Safety Issues in Current Practice

To represent a viable alternative to the approach to operational safety now embodied in current regulations and industry practices, a Reliability Program must focus on existing weaknesses. To develop system-specific insights, a study was undertaken to determine the risk-dominant attributes of the residual heat removal and reactor trip systems for the Browns Ferry Unit-1 plant from a review of existing PRA information for that plant, related LERs, and plant safety literature including emergency procedures. This allows focus on the parameters that govern the unavailabilities of representative, safety-related systems for nuclear plants so as to indicate the most important aspects of a reliability program. This work [12] confirmed, not surprisingly except perhaps for degree, the dominance of dependent (common cause) failures on risk-important sequences involving complex nuclear systems, highlighted the importance of the operator(s) being able to recover safety functions during an accident, and pointed to the need for plant-specific information to drive safety-related decision making.

These issues support the dependent-failure focus of a Reliability Program as called for above. It also supports the desirability of bringing the aforementioned risk/relaibility insights to prioritize accident management approaches. The need for plant-specific information for decision making calls for an in-plant performance monitoring and data collection system that can be readily tapped for safety-related evaluation.

A more general review of current operational and maintenance safety issues highlighting possible weaknesses was also undertaken. In this paper we'll limit the discussion to Technical Specifications. In August 1983, an

NRC Task Force on surveillance testing convened with industry committees established by the Owners Groups. Its findings [13] on Tech Spec content, shown in Table 1, support in part the results of a 1981 licensee poll [14] that characterized the surveillance requirements as "too much--too often" and asserted degradation of equipment, especially diesel generators, from surveillance testing alone. Many problems were cited, including plant transients initiated during surveillance testing. Less frequent testing was asserted to reduce challenges to safety systems. Some licensees, especially those not using Standard Technical Specifications (STS), stated that the compounding of test requirements that occurs whenever redundant trains of equipment need to be tested immediately following an unsuccessful test may be detrimental to safety.

From their findings, the Task Force made a number of recommendations for NRC and industry action which imply how a Reliability Program might be used to facilitate implementation of Tech Spec amendments. The thrust of the recommendations was to categorize systems in each of the four STS by risk importance. Starting with the high risk systems, reliability analysis models would be used to evaluate the frequencies and possible safety ramifications of surveillance tests. System test configuration and plant operational modes were to be evaluated to minimize unnecessary transients and shutdowns. Finally, the requirements were to be evaluated to minimize personnel time and radiation exposure. From such evaluations, Owners Groups would submit proposed STS changes and licensees would submit proposed changes in plant-specific Tech Specs to the NRC.

The Tech Specs safety issue supports the aforementioned recommendation for risk/reliability integration into operational requirements. A Reliability Program could use a reliability analysis capability to propose Tech Spec changes as shown above, but it could also use the <u>results</u> of a performance monitoring program as a measure of success (or failure). Allowing a speedup of the Tech Spec amendment process for licensees with a Reliability Program keyed to monitoring and then acting on system performance is discussed below under Regulatory Considerations.

Table 1. NRC Findings* on the Safety Impact of Technical Specifications

- Existing test frequencies and intervals may not be optimized. Frequent periodic testing of systems, with no compensating reduction in risk, results in unnecessary diversion of operators and other plant personnel, economic costs, and in some cases, excessive exposure to plant personnel.
- The action statements of some Specifications may cause unnecessary shutdowns that result in plant transients and challenges to safety systems. The primary concern was with those specifications that provide only 1 hour to restore equipment to meet LCOs.
- Particular types of testing may cause equipment degradation and progressively reduced system reliability. For example, fast cold start testing is likely to accelerate degradation of diesel generators. Further, cold starts are required only in the event of the large break LOCA with loss-of-offsite power, a low probability event. Degraded diesel generators cause the plant to be more vulnerable to the higher probability event, loss of all ac power.
- Allowable outage times (AOTs) are not based on the degree of system fault but rather on go, no-go criteria. AOTs that are too short cause unnecessary trips, transients, and fatigue cycling, and less thorough repair and post-repair testing before equipment is returned to service. AOTs that are too long increase risk.
- Sometimes increased surveillance testing of equipment in one train of a system is required if the other train is inoperable. This could (1) damage the redundant system, (2) place the system or plant in a more vulnerable mode, and (3) fail to return it to an operable condition. This concern applies to plant-specific Tech Specs rather than the STS.
- In the event of inoperable equipment, some action statements may require entry into a less safe plant model. Also, some tests may require placing the plant in a less safe configuration.
- The bases for Specifications generally do not provide explicit justification for the LCOs or the surveillance requirements. Thus, there is no baseline from which to assess the effect of a change and permit short-term emergency exceptions; also, there is little to inform the operator of the importance of the various requirements. Further, 10 CFR 50.59 requires an evaluation of the reduction in the margin of safety as defined in the bases for the Specifications to determine whether a proposed change, test, or experiment involves an unreviewed safety question.
- Certain surveillance requirements in radiation areas should be reviewed for overall impact on safety. Data obtained from 4 PWRs and 11 BWRs showed collective snubber-related doses ranging up to 220 person-rems per year at one of the facilities.
- Some Specifications unnecessarily consume the time and attention of plant personnel that could be used for purposes more important to safety. Also, they should not contain requirements for collecting information that does not add to operational safety.
- Some requirements vary among the STS without apparent reason, possibly penalizing some licensees and unbalancing safety for others.

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* U. S. Nuclear Regulatory Commission, "Technical Specifications--Enhancing the Technical Impact," NUREG-1024, November 1983.

Maintenance problems have also been identified as a major contributor to risk. An NSAC evaluation [15] found that maintenance was an important factor in the unavailability of the high-pressure coolant injection (HPCI) and the reactor core isolation cooling (RCIC) systems of BWR's. A major conclusion was that at least 40% of the HPCI/RCIC problems might have been averted by a high-quality preventative maintenance program. Discussions with personnel at plants with low HPCI/RCIC LER rates indicated that they have a formal and well-defined preventative maintenance program including records of maintenance performed on safety related components, review by others of critical component maintenance, and testing before the component is returned to service. While some of these reliability elements were present at all of the plants it appeared that the better plants (from a reliability standpoint) had successfully integrated them into a well defined maintenance program.

In a probabilistic evaluation of a 1300 MW Pressurized Water Reactor, maintenance was the largest contributor (about 50%) to the unavailability of the Emergency Core Cooling System (ECCS) following a LOCA [16]. This conclusion was found to be independent of ECCS design (e.g. number of trains) or LOCA break size with the proportional contribution nearly the same for the designs considered. Changes in the maintenance data (e.g., downtimes) did not have a significant impact on the "improvement factors" among the various design comparisons. However, maintenance improvements on component reliability were found to be more effective than redundancy for improving system reliability.

Other more general studies have identified the importance of maintenance in reactor safety. EPRI's review of nuclear power plant maintenance identified industry-wide inadequacies [17]. The Human Factors Society in a comprehensive evaluation of NRC human factor policies and practices also identified such inadequacies [18]. Pacific Northwest Laboratories (PNL) has developed recommendations for NRC actions related to maintenance [19]. PNL's general finding was that nuclear power plant maintenance had received far less emphasis than was warranted considering the potential impact on safety.

Using reliability evaluations to select maintenance strategies is the thrust of the reliability-centered-maintenance (RCM) practices in commercial airline, military, and aerospace industries. In the context of the approach herein, performance monitoring could flag problems and subsequent reliability evaluations could be used to select maintenance strategy changes. In short, the functions of the Reliability Program as developed below will be seen to facilitate an RCM approach.

Other Industries

Other industries were reviewed to identify reliability activities that could best be integrated with existing plant practices and NRC regulations to provide an alternative mechanism to current practice in maintaining an acceptable level of plant safety. Investigations [20,21] focused on NASA's Space Shuttle Main Engine Program, the U.S. Navy's Trident Missile Guidance System Program, FAA certification of airframe types, and FAA operation of ground control systems. The work was performed by Charles Stark Draper Laboratory (NASA, Navy, and FAA certification) and Reliability Technology Associates (FAA operations) under contract to ANL.

Selected conclusions are as follows: Institutional and mission differences make military and aerospace experience transferability tenuous. FAA type certification, which is somewhat analogous to the initial reactor plant licensing process, utilizes practices with features that might be applicable to the nuclear industry including the use of industry representatives who monitor and approve various production and manufacturing phases. However, these are related to new plant design and construction and therefore of minimal value in the current industry environment. FAA regulatory practices related to air carrier operation and maintenance have features that may be useful in the nuclear operations environment. In particular, the voluntarily established, FAA-approved reliability programs at the licensed air carriers which include failure experience collection and feedback, reliability monitoring, and the concept of reliability-centered maintenance (RCM) all seem to have significant applicability to the nuclear industry. More general practices of the FAA worthy of future investigation include certification of maintenance personnel, FAA/industry interactions through Maintenance Steering Groups and the anonymous reporting of the "Aviation Safety Reporting System."

Although this work indicated that the FAA/air-carrier-developed reliability activities had the potential to be of significant value, it is noted that the FAA was having significant problems in the implementation of its own approaches. A National Academy of Sciences study of the DC-10 crash over Chicago cited FAA failure to be aware of and to correct faulty maintenance practices; there is also the recent Eastern Airlines flight out of Miami which lost all its engines due to faulty metal chip detector maintenance practices which had been observed and recorded in the Eastern data system over 10 times in the previous year without FAA actions. These shortcomings point out the necessity to resist the tendency to charge headlong into implementing an analogous reliability program in the nuclear industry without fully understanding its potential limitations as well as its strengths.

Reliability Program Functions

In our most recent research [22] we have coalesced the previously discussed information to develop a viable Reliability Program model that focuses on identified operational safety issues. The next phase of this research is to test and refine the model via case studies addressing system/component-specific problems, in-plant demonstrations and value-impact analysis so that it could be cost-effectively used by licensees to assure that an adequate level of safety is maintained over plant life.

The three major functions of the Reliability Program developed herein are plant performance monitoring, the associated evaluation of this performance, and the integration of the necessary corrective actions into day-to-day operation. Simplistically the performance monitoring function tracks and trends the actual plant and system performance as well as related industry experience to identify existing or potential in-plant problems that affect safety; the evaluation function performs root-cause or related reliability analysis for the identified in-plant problems and determines the applicability of identified industry or generic problems; corrective action integration provides the mechanism to recommend actions based on the root cause and generic applicability analysis and incorporate these as necessary into day-to-day operations.

Depending on their scope, the performance monitoring function can provide system, component, or operations "performance signatures" that allow

-- online comparison with past performance, industry performance, and safety criteria
-- forecast of corrective action needs
-- a report card for corrective action success
-- input to plant risk and availability models,

while the performance evaluation function can provide

-- root cause or related reliablity analysis of in-plant problems or safety concerns
-- applicability analysis of generic or industry problems
-- justification of operational requirements changes in Tech Specs
-- input to selection of test and maintenance strategies in a reliability-centered-maintenance framework
-- risk-importance prioritized systems, components, and root-causes of failures.

The corrective action integration function is necessary to provide

-- sensibility/desirability of corrective actions
-- orientation of operational/maintenance staff to safety and reliability insights
-- management awareness

In concert, the above functions can be developed in a structured Reliability Program framework to incorporate the "repeating themes" of past recommendations and address the "safety issues" identified as dominant in both the system-specific and the general operational and maintenance safety reviews. The benefits to a licensee that accure for these functions, depend on the scope of the Reliability Program. A licensee may choose to implement a Reliability Program for a single system, such as emergency AC power, or for a class of components such as valves. Or he may choose to take a broader perspective and include all safety systems or risk-important systems, for example. Determining guidelines for the benefits of a Reliability Program as a function of scope is being evaluated in our research.

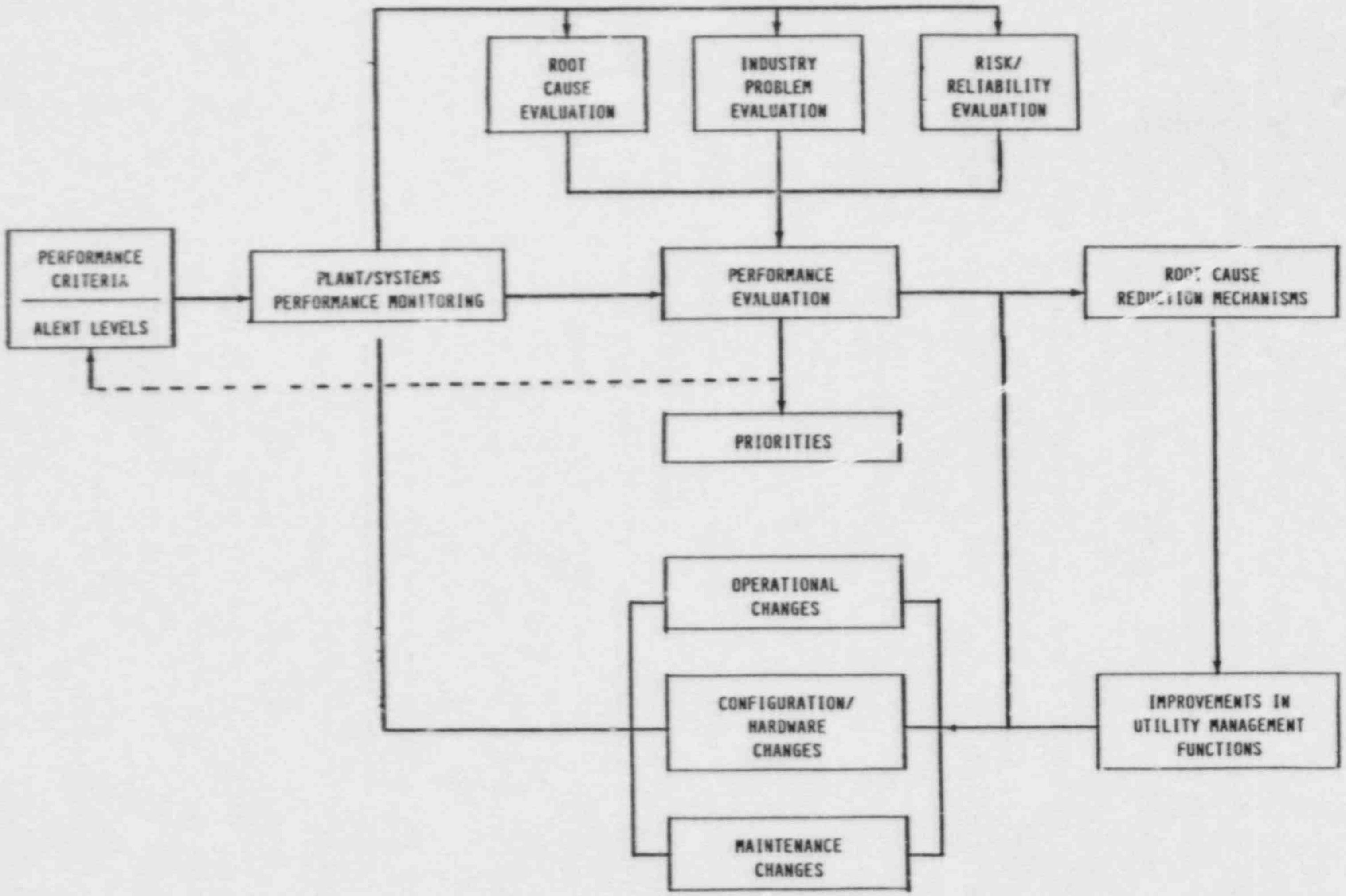
Supporting or interfacing activities include systems risk and reliability analysis to provide initial and updated-as-needed rankings of the importance of the various plant operations and systems so as to prioritize Reliability Program resources. The broader role of these analyses in evaluating dependent failures, Tech Specs and other test and maintenance requirements, as well as prioritizing other plant activities supporting safety has been discussed. These analyses also support the establishment of the proper performance measures and associated alert levels. Another activity is data collection and analysis to support the tracking and trending of in-plant operational safety reliability performance and related industry experience. Considerations for the scoping of these activities include the risk importance of the affected systems and operations as well as overall program mission. For example, these activities could also be used to optimize plant availability.

The general focus of the approach to operational safety envisioned herein is on determining the root causes of performance failures and taking the appropriate actions before serious problems ensue. This is illustrated in Fig. 1. First, it is envisioned that reliability performance criteria and associated alert levels be established by the licensee for systems or operations judged most important to risk. Although PRA techniques can be used to establish such priorities, the engineering judgement of licensee staff could also be used (e.g., a PRA is not needed to establish that the reactor trip system or the maintenance of its circuit breakers is important to risk). Monitoring and trending the performance of these systems or operations against these criteria is used to forecast potential serious failures or flag repeating cause categories of failures.

For serious non-performance, as measured by the penetration of the alerts, root-cause evaluation would be performed. Dependent failure evaluation should also be considered as indicated by the risk/reliability evaluation box of Fig. 1. Less serious non-performance, measured by a lesser penetration of alerts, would also be evaluated but not at the same level of detail. The performance evaluation is intended to result in identification of potential root-cause reduction mechanisms. The Corrective Action Integration Function enables the proper input of Operational or Maintenance staff in establishing selected corrective actions, especially where operational, configuration/hard-

Fig. 1

RELIABILITY PROGRAM PROBLEM APPROACH



ware, or maintenance changes appear warranted; it also enables the input of management if changes in utility management functions seem called for.

The arrow from Performance Evaluation to Priorities denotes the potential for reprioritizing the importance of systems, components, or selected operations at any point. The dotted line from Performance Evaluation to Performance Criteria denotes the potential for changing criteria or alerts that simply prove unworkable.

Major issues under study include prioritization of resources, that is identifying guidelines for establishing the proper coverage and level of performance monitoring evaluation for the safety systems and related operations. Risk-importance measures are being explored. Similarly the establishing of performance criteria, risk- or reliability-allocated standards, and associated alerts should have guidelines. Criteria for weighting the frequency, severity, or causes of incipient, degraded and catastrophic failures and synthesizing this information in system performance "signatures" are being investigated. The integration of the cited Reliability Program functions into existing practices should minimize impact on current operations and proper interfacing modes with current practices must be investigated. The developmental approach for all these issues is to perform case study applications on specific known industry problems, e.g., diesel generator unavailability and use trial demonstrations in participating plants to screen the most essential elements of a Reliability Program.

With respect to integration with current practices, it is important to acknowledge that all licensees have elements of the Reliability Program approach discussed here, some more than others. For example there is a wide spectrum of in-plant performance data collection and evaluation packages in-place [22]. Similarly, the evaluation capabilities of licensees vary over a wide spectrum. Hence, implementation benefits and costs will similarly vary.

Regulatory Considerations

Regulatory considerations are also important. Although the FAA approaches are not perfect, they do appear to have resulted in less adversary than is present in the nuclear environment. They also seem to provide commercial carriers with the incentive to improve reliability. Reliability programs are voluntary, but carriers without them are forced to revert to hard-time restrictions which carry stringent economic penalties. Analogously, utilities could have the option of staying with initially based Tech Specs or establishing RPs to allow them to take advantage of more reasonable requirements. For example, the standby nature of most safety important components requires that their state of readiness be measured via periodic tests that indicate their operational status. The frequency of these tests is established prior to operation and included in a plant's technical specifications as part of its operating license. As cited above these fixed tech specs sometimes include surveillance requirements which are inconsistent with reliability performance of the equipment under surveillance. This imbalance can impact risk by increasing the potential for test induced failure, and decreasing the time available for the plant staff to attend to issues of greater safety significance.

As discussed, this problem has been recognized by both the industry and the NRC (cf. Generic Letter 84-15) and has resulted in a significant number of industry requests for relief from individual requirements. Unfortunately because the requirements are part of the conditions under which a licensee has been granted an operating license each accepted request must result in the granting of an amendment to the operating license. Further since there is no established system or standard format for amendment request submittals each one must be considered by the NRC on an individual basis. The ad hoc nature of this process increases the potential for inconsistency in the decision making. Its case-by-case nature has overloaded the system with requests, often causing long delays in the implementation of tech spec amendments believed by both the licensees and the regulators to enhance plant safety.

If a mutually acceptable and practically implementable Reliability Program could be defined for the nuclear industry and if the Reliability Programs of individual utilities could be approved by the NRC and incorporated properly into the licensing process then these programs could alleviate this situation. That is, they would provide a mechanism for more tightly tying Tech Specs such as surveillance requirements to achieved performance and for more systematically and rapidly dealing with legitimate requests for such items as Allowed Outage Time (AOT) extensions, and increases in the surveillance Test Intervals (TI). It is the intention of this research to define and describe such a program.

Related Research

Much of the insights to be gleaned in refining the functions of a Reliability Program will come from reviews of other research. Table 2 presents a selected summary of ongoing NRC and industry initiatives and their potential interfaces with the Reliability Program research described herein.

Summary and Future Directions

In summary, the work described herein is part of a research project to develop and recommend a Program of coordinated reliability engineering and management approaches which could be applicable in the nuclear industry environment. It is intended that this Program could interface with on-going industry and regulatory programs to provide licensees and the NRC an alternative means for maintaining an acceptable level of nuclear power plant safety over the lifetime of the plant. It would establish reliability achievement levels for structures, systems, components, and operations compatible with the impact each has on overall plant safety. This would focus the attention of the program onto systems and components in accordance with their importance from a safety perspective, and establish reliability performance measures and corresponding acceptable performance levels consistent with the associated risk importance. The Program would further establish auditable criteria in each case to alert licensee management and the NRC to instances of unacceptable performance and provide for a pre-established systematic mechanism for determining and integrating appropriate corrective actions.

On-going and future work will develop a Reliability Program structure and associated activities for the complete lifecycle of a power plant. However since it is recognized that the primary emphasis in the next several years will be on operational safety rather than on new design, the near term work is directed toward developing a Reliability Program for the operations phase of a

Table 2. Selected NRC and Industry Research Programs Outlined for Review of Interfaces with the NRC Reliability Program.

- -- NRC's (RES/DET/EEBR) Nuclear Plant Aging Research (NPAR) with the thrust of the review being to factor in the findings of the following NPAR activities into corresponding RP activities
  - -- correlation of risk and aging trends
  - -- evaluation of operating experience and current surveillance/inspection technology including in-situ monitoring methods
  - -- identification of performance indicators.
- -- NRC's (RES) Technical Specifications (TS) Program with the thrust of the review being to factor in the results of their reliability assessment program and data development requirements activities. These activities relate to developing and demonstrating approaches/procedures for determining TS issues such as granting exemptions to allowed outage times (AOTs) or surveillance test intervals.
- -- NRC's (RES) Accident Sequence Evaluation Program (ASEP), with the thrust of the review being to determine the utility of ASEP results as a reliability model and data base for an RP.
- -- NRC's (RES) In-Plant Reliability Data System (IPRDS) with the thrust of the review being to factor in the findings of IPRDS activities to develop a comprehensive, component-specific data base for PRAs that would likely provide a model for the in-plant data base element of the RP.
- -- NRC's (RES) Accident Sequence Precursor (ASP) Study with the thrust of the review being to assess the applicability of ASP results for use as an information base for the RP.
- -- NRC's (RES) Risk-Importance Research with the thrust of the review to assess which importance measure parameters have potential for inclusion as system, component, and operations prioritizers for RP activities.
- -- NRC's (DHFS) Human Factors Programs with the thrust of the review to assess the applicability of people-oriented approaches (e.g., modifications to operator training or procedures), maintenance (M) modification activities (e.g., design for maintainability, M procedures and documentation, M personnel qualifications and training, preventative M, M work authorization and control, outage planning and management of inventory), the man machine interface question (e.g., role of simulators for training and their use in the control room), and human reliability data and human risk analysis and other human factor methodology for potential use in identifying RP interfaces with ongoing plant functions.
- -- NRCs (NRR) Maintenance and Surveillance Program with the thrust of the review being to explore the interface between the RP and Maintenance and identify how the RP can be used to prioritize and monitor maintenance activities and schedules.
- -- NRC/NASA/KSC's Systems Assurance Analysis (SAA) Program with the thrust of the review being to see how the FMEA and hazards analysis approach, including problem closure, fits into the Performance Evaluation Function of the Reliability Program.
- -- EPRI's Reliability Centered Maintenance (RCM) Research whose objective is to evaluate the cost effectiveness of this technique in an operating nuclear plant with the results directly applicable to the interfacing of an RP with maintenance.
- -- EPRI's Technical Specification program which complements NRC's program in this area and will perform case studies on actual operating systems technical specifications (e.g., diesel generator evaluation).

plant's life cycle. Program activities will be chosen to address system/component-specific problems such as weak performance in diesel generators supplying emergency AC power. The work scope includes in-plant demonstration studies which will be used to define the content of and to tailor the elements of a Reliability Program so they are consistent with the requirements of the NRC and yet responsive to what licensees feel is most warranted. Regulatory based value-impact analysis will also be utilized in conjunction with the demonstration studies to identify the most cost-effective ways of implementing a Reliability Program both from an industry and from a regulatory standpoint.

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INSPECTION PRIORITIZATION

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1. INTRODUCTION

Probabilistic risk assessments (PRAs) are analytical tools that use logic models of components and systems and probabilistic information to assess the overall risk of operating a plant. In the nuclear power industry, PRAs can be used to direct Nuclear Regulatory Commission (NRC) inspection personnel to activities that have the greatest potential to increase the safety of a nuclear power plant. The objectives of the Risk Assessment Applications to NRC Inspection Project, conducted at Oak Ridge National Laboratory, are to determine how the results of PRAs and other risk studies can be applied by NRC Inspection and Enforcement (IE) personnel and to determine what information must be combined with PRA results to make them most useful. Accomplishing these objectives will aid inspection personnel in prioritizing their limited resources and will increase their risk-limitation effectiveness.

2. APPROACH

A significant number of nuclear power plant PRAs, sponsored by both the NRC and the nuclear industry, have been completed in recent years. These PRAs contain a wealth of risk-based information on the plants that were evaluated; however, they were not developed with the intent of providing information for NRC inspection. To provide PRA information in a format that is useful to IE, we needed to first understand the relationship between inspection activities and plant risk.

Inspection resources are most effective in limiting plant risk through influencing the design and implementation of utility management programs. Utility management programs affect plant risk by the degree to which they control root causes of failure. Root causes of failure affect plant risk by influencing component failure probabilities, and component failure probabilities directly influence plant risk to the degree that the components are important to plant safety.

An example of a utility management program is a plant's maintenance program. The maintenance program should ensure that components and systems are properly maintained. A failure of the maintenance program (for instance, an incorrect procedure for maintaining motor-operated valves) could be a root cause of a single component failure or several component failures throughout the plant. Thus, when considering the relationships between inspection actions and plant risk, the factors involved include 1) inspection actions, 2) utility management programs, 3) root causes of component failures, and 4) component and system failure information in the PRA.

We developed a four-step approach for establishing formal relationships between inspection actions and plant risk. Figure 1 is a flow chart of

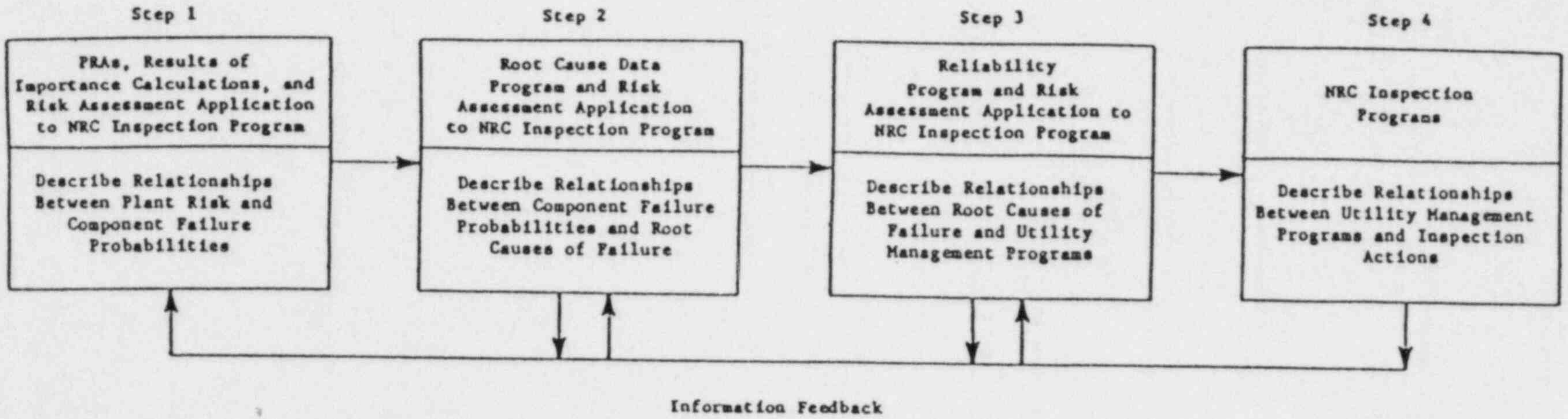

Figure 1 Proposed Steps in Establishing Formal Relationships Between Inspection Actions and Plant Risk

that approach. The upper portion of each box in Figure 1 details NRC programs and other information sources that provide the information necessary to perform the step described in the lower portion of the box. The arrows in Figure 1 show information flow paths among the four steps. Our current efforts to provide information to the inspector from each of the steps are discussed below.

<u>Step 1</u> **Describe the relationships between plant risk and component and system failure probabilities.**

The relationships between plant risk and component and system failure probabilities can be quantitatively described using risk importance measures. Many such measures exist,[1,2,3] and different importance measures provide different component and system rankings. To avoid providing misleading information to the inspector, the importance measure must be matched with the inspection activity. Because most existing importance measures were developed without regard to their use for prioritizing inspection activities, it is difficult to define the appropriate matches between inspection actions and importance measures. We are coordinating our efforts with the Risk Measures Program being performed at Battelle Columbus Laboratories (BCL).

<u>Step 2</u> **Describe relationships between component and system failure probabilities and root causes of failure.**

The relationship between component and system failure probabilities and root causes of failure (i.e., a root cause type) can also be quantitatively described using risk importance measures. It is important to the IE Prioritization Program to be able to determine the integrated effect (i.e., importances) of a root cause of failure on plant risk because:

1. A given root cause of failure can affect many (or all) component failure probabilities for the components considered in the PRA of the plant.

2. Properly designed and implemented utility management programs are the most effective way to globally limit or reduce the contribution a root cause of failure makes to the affected component failure probabilities.

3. Inspection resources are used most effectively to improve the design or implementation of utility management programs, which in turn affect many components by controlling root causes of failure.

A root cause type may be responsible for many root cause events. For example, the root cause "improper maintenance" may result in the root cause event "failure to tighten a bolt on a pump." Some root cause events can result in multiple component failures while other root cause events affect only a single component. We have developed a methodology to determine the combined effects of all root cause events induced by a given root cause. The following is a list of root causes we plan to evaluate using this methodology:

1. design inadequacy
2. faulty manufacture
3. improper installation
4. improper testing
5. improper calibration
6. improper maintenance
7. improper configuration control
8. improper operation
9. aging (wear out)
10. harsh environments beyond component design bases
11. other causes (random failures and failures with undetermined causes)

If the PRA being used has appropriately modeled both independent and dependent effects of root causes, then the methodology properly accounts for 1) root cause events that affect multiple components simultaneously to produce dependent failure events and 2) root cause events that independently affect several component failures.

Unfortunately, most presently available PRAs have not modeled either the independent or dependent effects of root causes of failure. However, it is possible to use PRA results to determine the independent effects of root causes of failure, and, if the PRA included a dependent failure analysis, it is possible to determine dependent effects of root causes of failure from the results of the dependent failure analysis.

To evaluate the independent effects of root causes of failure this method requires that root cause fractions be identified. A root cause fraction is defined as the fraction of an event's failure probability that is attributed to a particular root cause. The root cause fractions are being determined by the Root Cause Data program, a portion of the RES Data program.

NRC's Risk Methods Integration and Evaluation Program (RMIEP) is performing a new type of PRA that will include analysis of root causes of failure. The inclusion of root cause analysis in PRAs bridges the gap between the component failures and functional areas dealt with by utility management and NRC inspection programs. Such PRAs will be much more useful for evaluating the effects of plant management and inspection activities on root causes of failure and, therefore, on plant risk. If past PRAs are to be more useful documents for reliability and inspection programs, they should be updated to include analyses of root causes of component failure.

Step 3 Describe the relationships between root causes of failure and utility management programs.

The relationships between root causes of failure and utility management programs are being described qualitatively because there is no hard data available for quantitatively modeling these relationships. These relationships are described by the activities a utility performs that influence the occurrence of root cause events in a plant.

The utility responsible for operating a nuclear power plant must establish a number of in-house management programs to comply with various licensing requirements. These programs affect plant safety as well as plant operation. They help prevent the occurrence of root causes of failure in the plant. Because a root cause of failure can result in the failure of more than one system or component, eliminating or minimizing the occurrence of root causes is very important. For example, a maintenance program with incorrect procedures can adversely affect many components. The utility (independently or through the inspector), by identifying and rectifying such problems, can significantly limit plant risk.

This project has identified the following reference set of utility management functions and investigated methods for assessing the roles these functions play in reducing the frequencies of root causes of failure.*

- Administrative Control
- Calibration
- Design Control
- Emergency Planning
- Fire Prevention/Protection
- Health Physics
- Housekeeping
- Inventory Control
- Maintenance
- Operations
- Procurement
- Quality Assurance
- Security
- Surveillance, Testing, and Inspection
- Training

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*Management functions, instead of specific utility programs, were identified because some functions are managed by programs with titles that vary from utility to utility.

Licensee event reports (LERs) are being reviewed as part of this project to establish the relationships between management functions and root causes of failure contributing to reportable events. (A similar review successfully determined the reasons for variations in the number of LERs that involved the high pressure injection systems for several different boiling water reactors.[4]) The results of such reviews provide information useful in determining which management functions influence the occurrence of root cause events.

In this project, the qualitative relationships between management functions and root causes of failure are being modeled using influence diagrams. Direct management function influences on root causes of failure are evident since many management functions specifically address individual root causes. Indirect management influences arise from interdependencies among management functions.

Step 4 Describe the relationships between utility management programs and inspection actions.

The relationship between utility management programs and inspection actions is best understood by inspection personnel. The inspector's accomplishing changes in utility management programs is frequently a complicated process, and the personalities of utility management and inspection personnel are factors important to the outcome of attempts to change utility management programs. IE personnel are familiar with the plant and with their interactions with plant management. Therefore, they are in the best position to assess 1) whether performing an inspection activity at a particular plant will have the desired risk impact and 2) what resources are required to accomplish that risk impact.

The information provided by Steps 1, 2, and 3 will help IE personnel identify the potential risk impact of performing various activities;

therefore, IE personnel should use this information in conjunction with their understanding of the facilities when making decisions on how to best apply their resources.

3. CONCLUSION

The allocation of inspection resources is currently performed without the benefit of information that relates the results of a plant risk assessment to the objective of inspection tasks. The Risk Assessment Application to the NRC Inspection Program aims to provide IE this risk-based information. This information can be used to allocate limited manpower to in-plant inspection activities that have the greatest impact on plant risk.

Future methods development work on this project will focus on the first three steps of the four-step procedure presented. Future applications work will concentrate on using the four-step procedure to supply IE and the regions with PRA-based information in areas recommended by inspection personnel.

While it is believed that this four-step procedure will provide useful information to NRC inspectors, results from this procedure cannot be obtained until 1) results are obtained from RMIEP-type PRAs where the analysis is extended to the root cause level and 2) appropriate importance measures are obtained from the Risk Importance Program. Also, root cause fraction information is needed from the Root Cause Data Program. In the interim, we are focusing a portion of our current effort on providing insights from the PRA that can be immediately useful to NRC decision-makers. These short-term results, along with advances made with the longer term four-step approach, will be documented in a March technical progress report.

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REACTOR SAFETY RESEARCH: VISIBLE DEMONSTRATIONS AND CREDIBLE COMPUTATIONS

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INTRODUCTION

When the Wright brothers first flew an aircraft successfully, they did not have the slightest idea that their pioneering endeavors would result in a monumental space research program and commercial aviation industry some day--let alone dreaming the possibility of landing an aircraft with 250 passengers aboard using automated control options.

When Enrico Fermi and his fellow scientists demonstrated a self-sustaining nuclear chain reaction in a laboratory, they were already dreaming of the possiblity of generating the electricity more economically some day.

Typical of all the major scientific breakthroughs both the aviation and nuclear research programs received considerable attention and support from the government in their embryonic stages. Government resources helped in developing military aircraft and the associated technology. Government support in nuclear research was also profound for propulsion and weapons and their technology. The technologies were the basis for embarking on commercial enterprises in aviation and nuclear electricity generation.

The industries have many similarities in terms of missions, operations and regulations. The major differences between the two industries are summarized in Appendix A. It is interesting that both industries are regulated and use sophisticated technologies. However, the airline industry enjoys the overwhelming public acceptance and support; and the nuclear industry does not, now.

In adapting to rapid changes in technology, the airline industry has taken advantage of fast-moving research to improve the reliability and performance goals. In the nuclear arena we see other countries seem to be more aggressive than the USA on incorporating fast moving technology for availability and safety improvements[1,2].

In spite of a number of airline accidents, the Wright brothers' adventurous endeavors and the following developments have established the airline industry in "harmony" with the public. The achievements of the pioneering nuclear scientists and their followers have not achieved a similar degree of harmony between the public and the maturing nuclear industry. The airline "harmony" has even stood the test of the trial of the supersonic transport which was completed or expeditiously dispatched over a rather short period of time.

As researchers, we could ask ourselves a question related to the above observations:

Q. Has the growing base of nuclear research findings, and the associated computations, beneficially supported harmony between the public and the nuclear power institutions?

Having posed the question, we should attempt to answer objectively:

A. The research and its findings have probably not benefically supported the desired harmony. The maturity in the nuclear technology has grown very rapidly, perhaps more rapidly than can be appreciated easily by non-specialist and non-technical groups.

We have not been very effective in communication on our research results. For example, visible demonstrations of complex phenomena can be used to convey understanding and assurances on nuclear safety concerns.

An example supporting the contention that we do not present the fruits of our research well will help. A number of years ago people were, as they still are, studying the behavior of fuel under irradiation. We remember two organizations in that field whose reporting practices differed markedly. One placed most emphasis on those findings that suggested deficiencies in performance. The other placed most emphasis on demonstration of performance. Both organizations generally had similar findings but the reporting perspectives were different. This is the classic case of a glass that is half full vs. one that is half empty.

Many of the current safety research projects at the Electric Power Research Institute (EPRI) are described in the following sections. The question posed above played some role in their evolution with regard to topical focus and manner of procurement of the technical information.

EPRI RESEARCH

The Electric Power Research Institute (EPRI) has been conducting nuclear safety research for a number of years with the primary goal of assuring the safety and reliability of the nuclear plants by visibly demonstrating the existence of quantified safety margins. The visibility has been emphasized by sponsoring or participating in large scale test demonstrations to credibly support the complex computations that are the basis for quantification.

The visibility of a large scale demonstration tends to enhance the harmony cited earlier since the non-specialists can understand and appreciate what has been obtained, particularly when it involves difficult and incomprehensible computations.

A past example of this visibility are our completed turbine missile projects. Clearly, there was a concern that a portion of a turbine disc flying from a turbine might penetrate a containment building wall. The work before the EPRI tests were based on uncertain computations. The EPRI program focused on building walls with concrete reinforcement. Real turbine segments were accelerated to postulated speeds and allowed to impact on the walls. Four things came out of the program:

1. The depth of penetration of the discs into the walls upon impact were more or less consistent with prior predictions, (2 ft. into a 5 ft. thick wall).

2. The extent of backface spalling (cracking in the wall) was significantly less than prior predictions (e.g., without steel liners) .

3. Questions about "Peninsular" versus "Parallel" turbine placements could be considered on the basis of observations and,

4. The conclusions above were credible because even the least sophisticated "Commentator" could see and identify with the result.

The general approach to technical quantification along with visible and easy identification has been a primary motivation on many of the EPRI safety research efforts. The success has not been uniformly good but it is encouraging. As we look toward closing out safety issues, the pressure is on us as researchers to accelerate our efforts to communicate the substance of our accomplishments, particularly to the audience of non-specialists, in terms that they can understand. We think "we've got the goods". We also think we have not made the extraordinary efforts to effectively demonstrate what we now have and what we will have. This is where we can all learn from the airline industry. In the following sections we will briefly summarize highlights of current activities. Hopefully, these summaries will be in the spirit of the discussion above. If not, feedback would be appreciated.

SOURCE TERM

"Source term" is the terminology used to describe the type, quantity and timing of the radioactive fission product release from the containment building to the environment during postulated low-probability reactor accidents. Generally, the source terms are 'conservatively' evaluated from the consequences of these accidents in studies such as WASH-1400. Assuming high source terms gives rise to predictions of a class of severe consequences of the low-probability events. These in turn, can and do lead to public emergency planning group reactions. Admittedly, nuclear researchers have not focused on this area until recently. The only way to arrive at realistic and credible estimates in the studies is to generate data to validate the methodologies used for consequence analysis.

In a recent talk (3), two accidents were citied which occurred in military reactors; each resulted in reactor damage, with insignificant amounts of radioactivity released to the environment. The only commercial reactor accident at Three Mile Island, Unit 2 (TMI-2), in addition illustrated very clearly the fact that the source term is calculated in a highly conservative way. The release of 15 CI of ^{131}I to the environment from TMI-2 was less than 1/100,000th of the predicted amount using the theoretical models. Although xenon and krypton were released, they were quickly dissipated. The TMI-2 accident demonstrated that the radioactive fission products remained substantially within the reactor building structures. These findings are extremely encouraging. EPRI is continuing its research efforts in the 'source term' area recognizing the potential impact it may have in changing the public perception. Many other foreign countries, NRC, DOE and industry groups are also actively pursuing research in this area. There is a presentation today by Dr. Richard Vogel, et al. (4) on EPRI experimental and analytical research in this important area. (5) The major experimental works are summarized in Table 1 and the code development work is described in Table 2. The major research efforts are summarized below:

STEP (Source Term Experimental Program)

EPRI is working with Ontario Hydro, Belgonucleaire, NRC and DOE to collect data under prototypic severe accident conditions using the TREAT reactor facility. The experimental system consists of a bundle of four pre-irradiated, Zircoloy- clad LWR fuel pins contained in a capsule-type vehicle made up of a cylindrical pressure vessel, a highly insulated inert flow channel with a steam source and sink and a source term measurement system. The part of a hypothetical degraded core accident to be simulated is from fuel exposure to steam through volatile fission product release. Chemical and physical characterization of volatile fission products will be attempted with spacial and temporal resolution achieved by disbursed sample tabs and sequential aerosol sampling with post-test analyses. The first of four scheduled tests has been completed with test information being analyzed at this time[6].

Fuel Debris Bed Coolability

TMI-2 demonstrated that a bed of core debris can be effectively cooled by water and steam during a severe accident. Under EPRI sponsorship, researchers at UCLA and University of California at Berkeley are studying the coolability parameters of a self-heating particle bed under pool boiling and forced-flow conditions and the quenching of such a hot debris bed by top or bottom flooding. Our tentative conclusion is that, given an adequate water supply, a fuel debris bed of large particles (~1 cm) can be cooled either in or outside the vessel[6].

Large Scale Fission Product Transport Testing (Marviken and LACE Experiments)

EPRI and NRC are working with eight countries in an approximately full-scale series of tests at Marviken, Sweden to study the

Table 1: Source Term Major Experimental Projects

		Location	Cost	Sponsor/Date	Objective	Results to Date/Expected
1.	STEP (Source Term Experimental Program)	ANL's TREAT reactor, Idaho	$4M	Ontario Hydro Belgonucleaire USNRC DOE EPRI (Summer '84)	Characterize volatile fission product release from a deteriorating fuel element to understand what species are released, including their chemical and physical form.	First test completed. Results being analyzed.
2.	Marviken Project	Marviken, Sweden	$10M	8 foreign countries EPRI NRC	Full-scale decommissioned nuclear plant is used to quantify what PCS retention factors can be reliably counted on during an accident.	Aerosol particle sizes seen were larger than expected.
3.	Water Scrubbing Experiments	Battelle Columbus Labs		EPRI	To validate a methodology for calculating realistic pool scrubbing decontamination factors of fission products.	Results will provide the data base for quantifying the source term (attenuation factors in water close to boiling temperatures).
4.	LWR Containment Aerosol Experiments (LACE)	HEDL	$6M	EPRI	To "realistically" represent the true prevailing conditions in the damaged reactor containment building and to reduce the conservatism introduced by the steady-state assumptions made in hypothetical accident analyses.	Large-scale aerosol air-cleaning scoping experiments were run.
5.	Containment Experiments	Portland Cement Institute	$3.1M	EPRI	To verify the integrity of the containment structure during the hypothetical accident conditions under loads beyond design loads.	Buildings are a lot tougher than the conditions they were designed for.

TABLE 2: EPRI Source Term Computer Code Development and Applications

1.	CORMLT	Models heat up, liquefaction and slumping of a PWR core and structural materials from the core uncovery until the time the core collapses into the bottom head; core geometry changes during accident progression can be modeled. Flows and temperatures in upper plenum and Primary Coolant System (PCS) are calculated.
2.	PSAAC	Models thermal-hydraulic processes in the BWR and PWR primary coolant systems predicting time variations of the gas temperatures, structural temperatures, the gas flow rates, the condensation rates and the heat and mass transfer boundary layer parameters. Output of this code is provided as inputs to aerosol transport codes, which estimate fission product and core material retention within the primary system.
3.	SUPRA	Calculates the time-dependent decontamination factor of fission product aerosols passing through water pools; also calculates the pool conditions and containment atmosphere conditions.
4.	RAFT	This Reactor Aerosol Formation and Transport (RAFT) code predicts the size distribution and composition of aerosols formed from condensation of volatile fission product and non-radioactive control rod materials released in postulated LWR accidents.

attenuation of aerosols in the primary coolant system. The basic PWR arrangement of the facility allows aereosol transport through the pressurizer and relief tanks. The facility will also be modified to provide a flow path of interest to BWRs. The Marviken experiements to date show that (1) significant aerosol retention in the PCS is possible when condensing conditions or water pools are present, and (2) aerosol particles can agglomorate to large size, increasing their deposition rate.

The LWR Aerosol Containment Experiments (LACE) are investigating aerosol processes such as steam condensation and their effectiveness in removing the aerosols from the containment and auxiliary building. Containment bypass scenarious (such as PWR V sequence) are low in probability and high in risk. Thus LACE will pay particular attention to the passage of aerosols through a long pipe (such as in V) and corresponsing simulated auxilliary building behavior. This program will clarify the currently most significant risk contributors to off-site consequences in a severe accident. The LACE experiments will be performed at Hanford Engineering Development Laboratory's Containment Systems Test Facility (CSTF). The attention of radionuclides escaping the primary system into the containment will also be investigated in this project[6].

Analytical Support

Many available NRC, IDCOR and EPRI computer programs are being used to analyze the experimental work that has been and will be done. In addition, development and validation of a limited number of new computer modules and codes, based on the test data will provide credibility to consequence analysis. The Reactor Aerosol Formation and Transport (RAFT) Code, in particular, is being used to predict the size distribution and composition of aerosols formed from condensation of a volatile fission product and non-radioactive control rod materials released in the accidents. Predictions using RAFT of the MARVIKEN tests to date agree with experimental data.

Water Scrubbing of Fission Products

WASH-1400 assumed a fission product decontamination factor (DF) of 100 for unsaturated water pools and of 1 for steam saturated water pools. It was surmised that higher DFs are frequently and usually encountered under the condition of concern. EPRI sponsored related experimental and analytical studies. The experimental program consists of three phases: Phase I uses single orifices in water; Phase II tests use multiple orifice configurations; and Phase III involves BWR downcomer and vent configurations.

Each phase consists of two parts: hydrodynamic tests and pool-scrubbing tests with fission product aerosols. The Phase I scrubbing tests have been completed. The single orifice Phase I scrubbing tests in subcooled pools show that steam mass fraction and aerosol particle size are the most sensitive parameters (Figure 1).

EPRI has developed a computer code called SUPRA (Suppression Pool Retention Analysis), which describes the scrubbing of fission products in water pools. SUPRA divides the scrubbing analysis into

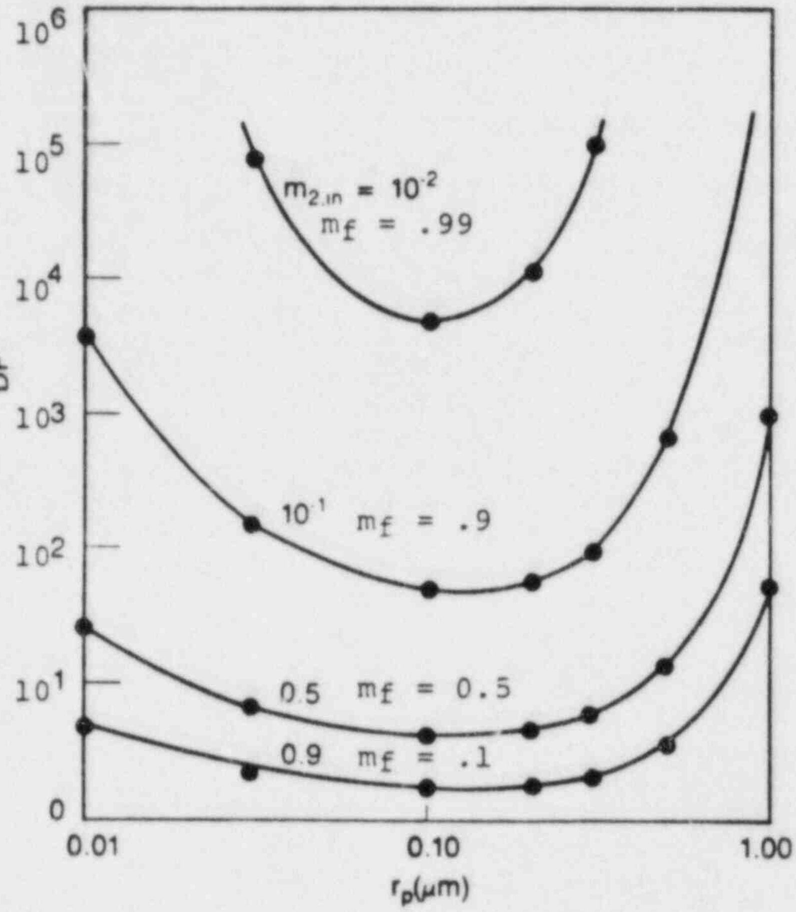


FIGURE 1. Effect of particle size(r_p) and steam mass fraction on DF as calculated by SUPRA. 3.5 meter deep pool with pool temperature 300K, inlet mass flow rate of 1.5 kg/s for various mass fractions of steam (m_f) and hydrogen temperature of 600K.

four zones: the injection zone, the mid-pool zone, the pool surface, and pool containment. SUPRA calculations have been compared to the pool scrubbing experiments. The comparisions have turned out well. SURPA was used to calculate the effect of particle size which is quite significant and mass fraction of steam on DF. These results are shown in Figure 1. In general, the experimental evidence suggests that sizeable particles are common with sizeable decontamination factors; larger than "classically" assumed.

Hydrogen in Reactor Containment

During a degraded core accident, Zircaloy can react with water to form hydrogen. This hydrogen can eventually be released to the reactor containment building. Combustion of hydrogen from the Zircaloy-water reaction represents a potential hazard and may effect the performance of the safety equipment. Also, the combustion of hydrogen released to containment may affect retention and resuspension of aerosols.

Results of EPRI's active research efforts from 1981 to date have provided a much better understanding of potential hydrogen combustion behavior. The threat posed by hydrogen to the containment building is less than previously thought. Plume speeds are very low, on the order of a few meters per second, far lower than sonic velocities required for detonation. The accident simulation tests showed a continuous burning without severe pressurizations (7,8,9). This indicates, the release of aerosols from the containment due to hydrogen combustion is seen to be very improbable.

Summary of Source Term Investigations

Source term experiments and analysis have produced very encouraging results to date:

- There is little, if any elemental iodine released during severe accidents. It is believed that iodine forms a salt compound CsI that is soluble in water and does not form a gaseous species that can migrate large distances.

- Water is extremely effective in removing hydroscopic fission products and this indicates that a 'low' source term is favorable under postulated LWR accident conditions.

- Based on the Marviken tests, radioactive aerosol particles are likely to be larger than now assumed. Moreover, they are found to be liquid, hydroscopic droplets further enhancing their retention.

- Pool Scrubbing is an effective means of removing fission products during hypothetical severe accidents.

The above results suggest a significantly lower source term than was used in the WASH-1400 analyses. The conclusions drawn from the recent evaluations by IDCOR, NRC and EPRI to re-evaluate WASH-1400 source terms is that the more we know about consequences of severe accidents the lower the source terms tend to get. Assuming that present results will stand the test of time, we may conclude that many of the current safety concerns will focus more on protection of plant investment. Off-site consequences are expected to be so low that they will cause minimal risk to the public and obviate need for emergency response planning.

SEISMIC RESEARCH

Seismic hazard is currently a public safety issue and seismic design has become a concern in construction cost and licensing of many nuclear power plants. The nuclear industry has recognized this and has taken steps to provide visible methods and data to facilitate stable licensing procedures. This includes focused research on seismicity and validated seismic design practices.

EPRI is currently taking steps toward establishing a Seismic Center whose major objective is to catalyze industry research and provide a technical basis for follow-up actions.

The Seismic Center activities will be in the following areas:

- Strong Earthquake Ground Motion Simulation
- Seismic Hazard Evaluation
- Soil Stability and Soil-Structure Interaction
- Prediction of Floor Response Spectra
- Structural Seismic Design Criteria
- Equipment Seismic Design Criteria
- Structure Equipment Capacity

Major recent efforts in three of these areas; earth sciences, geotechnical engineering and structural engineering, are summarized below.

Earth Sciences

The major Earth Science activity in the 1984/85 time frame is to generate a data base and develop a methodology for estimating the seismic hazard at nuclear sites within the Eastern United States. This program is expected to result in a comprehensive probabilistic seismic hazard methodology during the first quarter of 1985.

A modest research effort to address deterministic methods for earthquake ground motion estimation is also being implemented.

Geotechnical Engineering

The primary activity in Geotechnical Engineering is an investigation of nonlinear soil-structure interaction (SSI) behavior and, hence, a quantification of the conservatisms embedded in the current equivalent linear approach under strong earthquake conditions. EPRI has produced a nonlinear computer code, STEALTH, (10,11) which can be used for analysis of SSI. This code is available through the Electric Power Software Center (EPSC). The prime need in SSI is experimental data. In the past, EPRI has subjected a series of scale models to simulate earthquakes by using high explosives in New Mexico and New York (12,13).

To supplement the simulated earthquakes, EPRI, with the cooperation of the Taiwan Power Company, is constructing a 1/4-scale and a 1/12-scale model containment in Lotung, Taiwan. The location is a seismically active one, in which a large array of strong-motion accelerometers installed under the sponsorship of the U.S. National Science Foundation exists (Figure 2). Both the containment and internal components will be instrumented, together with the free field ground motions. The U.S. NRC will also participate in the Lotung project by sponsoring forced vibrations tests on the scaled containment models.

The Lotung site in Taiwan is of soft soil which is of particular importance in view of the strong soil-structure interaction (SSI) in this environment.

Structural Engineering

The primary focus in Structural Engineering is the quantification of the conservatism embedded in the current linear approach to piping system design.

With the collaboration of Consolidated Edison Company of New York, Inc., EPRI sponsored an in-situ piping test project at Indian Point-1 nuclear station [14,15]. An 8-inch diameter (20-cm) feedwater line inside the reactor containment vessel was used. Objectives were to obtain knowledge about the dynamic behavior of the piping system and realistic estimates of system response parameters, such as damping, pipe-support interaction, and system non-linearity. A major result of finding that average measured damping values for the system were 2-4%, which is consistently higher than the 1% value specified in NRC Regulatory Guide 1.61 for an 8-inch (20-cm) line at OBE levels (Figure 3). EPRI has submitted the data that support a more realistic damping guideline for nuclear piping design and analysis [16].

EPRI sponsored a nonlinear computer code development, ABAQUS- ND, for analysis of piping systems subjected to both static and dynamic loads. In recent years, this code has been expanded to become a general purpose code, ABAQUS-EPGEN, with co-sponsorship of Hibbitt, Karlsson and Sorenson, Inc[17-19].

EPRI and NRC have jointly sponsored a series of high magnitude piping dynamic tests at ANCO Engineers' test laboratory[20] to provide non-linear piping response data. Test specimens include simple two-dimensional, Z-shaped piping (funded by EPRI) as well as more complex three-dimensional pipings with and without branch lines (funded by EPRI/NRC). Results shown in Figure 4 indicate that large dynamic margin exists in piping and supports when compared to ASME allowable stress limit (Level D).

Toward developing a basis for simplified supports, Work is sponsored at UC Berkeley to investigate the use of simple cantilever-type ductile restraint for piping support[21,22]. EPRI has also contracted with R. Cloud and Associates to assess the use of box frame passive support for nuclear piping design application.

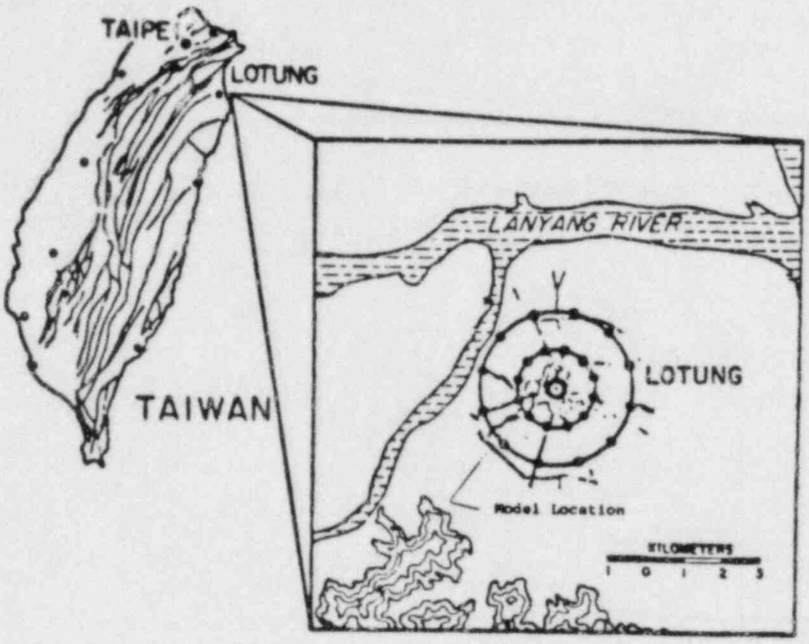


FIGURE 2a. Strong Motion Array

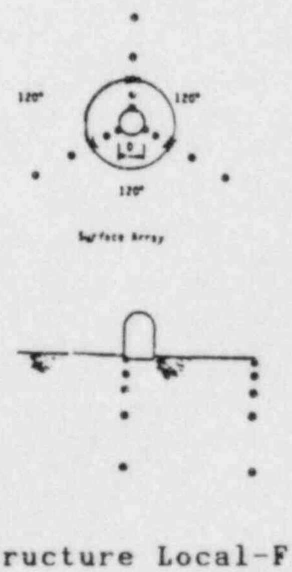


FIGURE 2b. Free-Field and Structure Local-Field Instrumentation

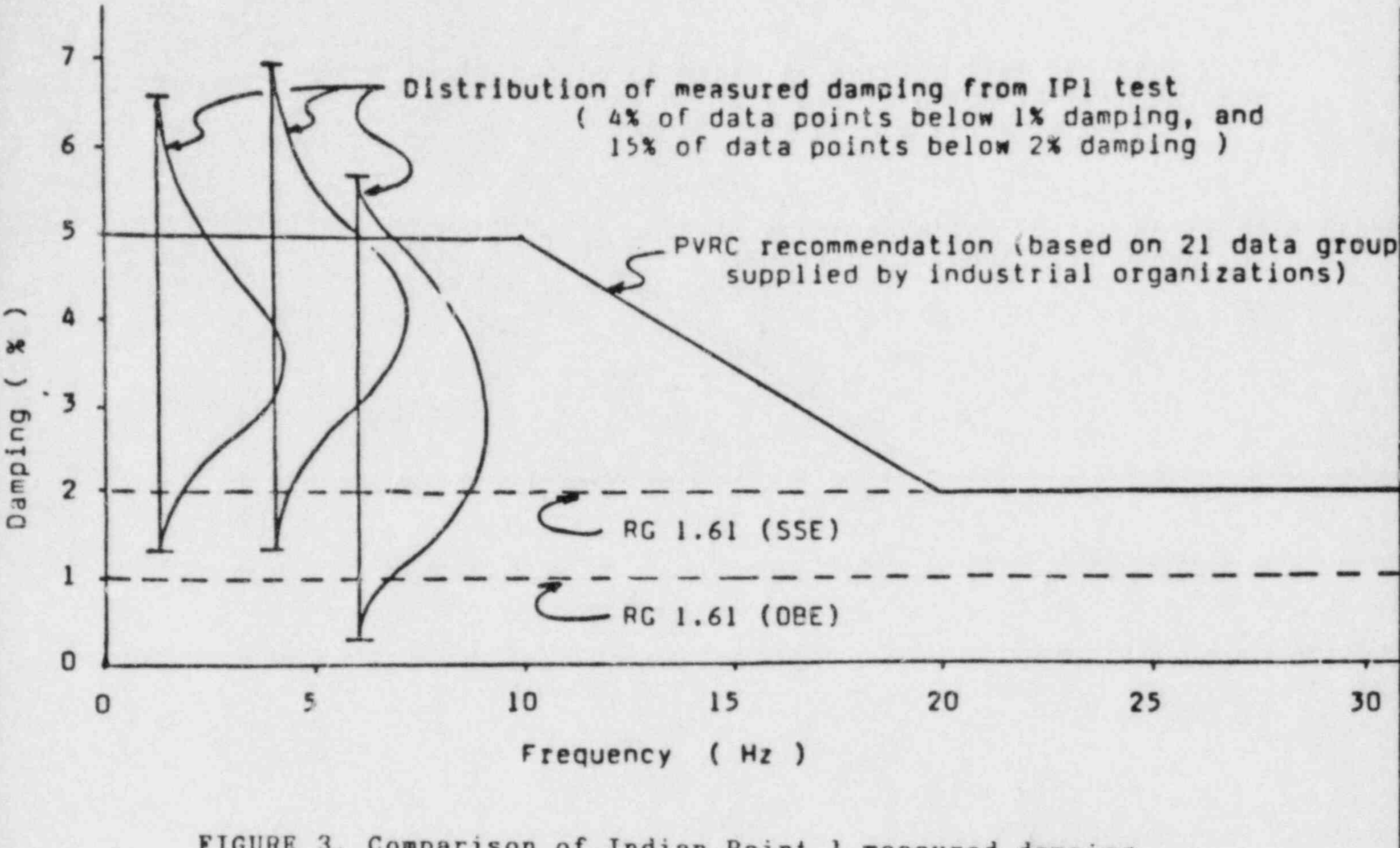


FIGURE 3. Comparison of Indian Point 1 measured damping against USNRC Regulatory Guide 1.61 value and PVRC recommendations.

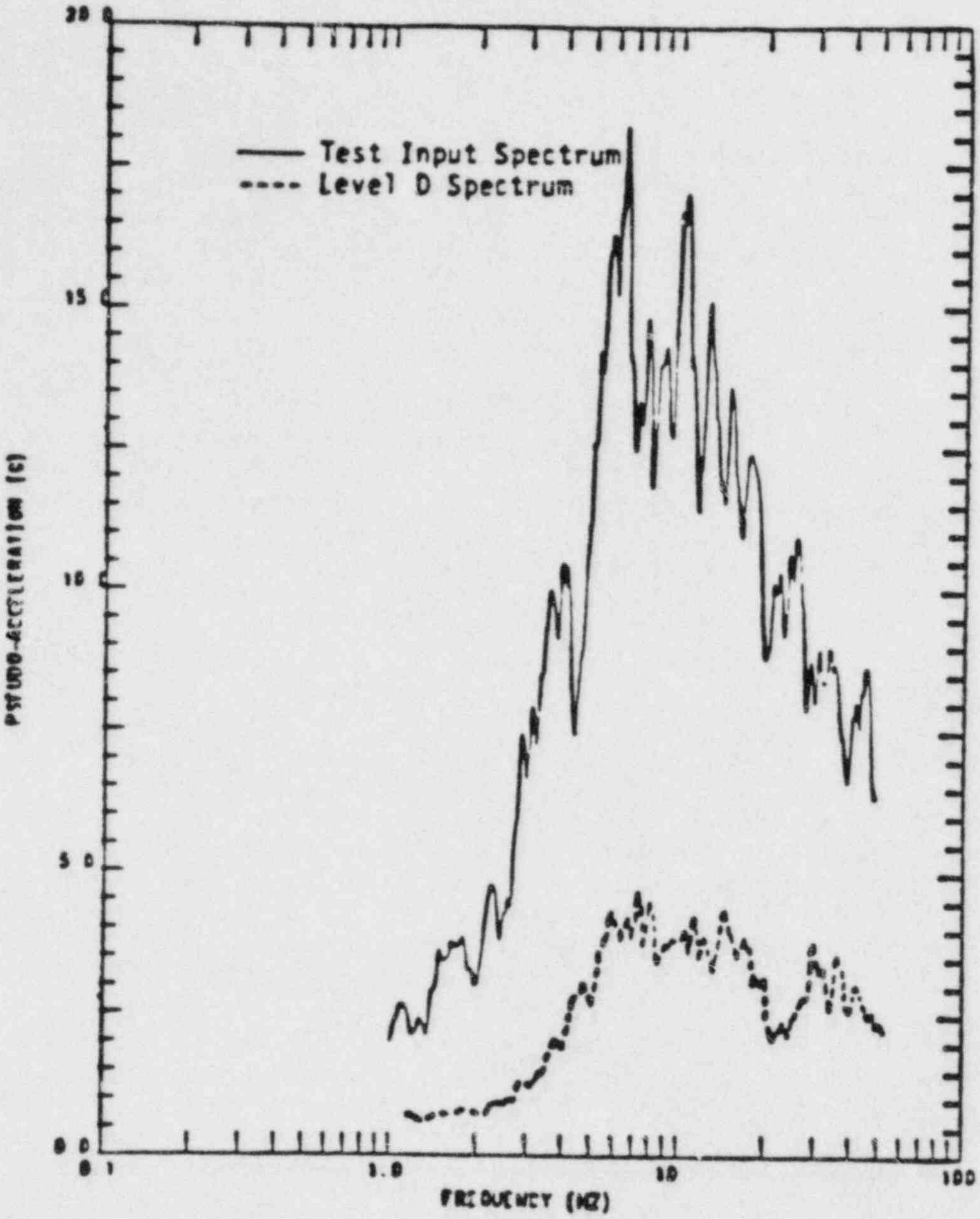


FIGURE 4. Response spectra comparison for high magnitude piping dynamic tests.

Concrete Containment Structural Integrity Program

This research is aimed at providing experimental data base and test-validated method for assessing concrete containment capability under postulated degraded core scenario. The research is being carried out at the Portland Cement Association on testing large - or full-scale structural elements of containment buildings. The analytical work is being performed by Anatech Corporation to develop and validate models and methods for concrete containment behavior analysis. The program is conducted in two phases. The first phase research is completed. It consisted of uniaxial and biaxial tests and analysis of simple 5-foot square by 2-foot thick reinforced concrete panels. Results showed maximum strain reached to be on the order of 2% and spacing of through-cracks ranged from 18 to 24 inches with maximum crack width approximately 0.35 inches. The second phase program is underway to test full-thickness (3.5 feet) containment segments with variety of prototypical design conditions, including penetration, discontinuity, and temperature effects. EPRI sponsored ABAQUAS code is being used for analytical correlation and model improvement.

The various seismic research projects at EPRI are summarized on Table 3.

SYSTEMS ANALYSIS AND PRA

The Reactor Safety Study, WASH-1400, showed that judgement alone is not a good guide to plant features and plant operations that are most important to safety. The formal licensing process deals with the large break loss of coolant accident and single point failures. The Study showed that small break LOCA, and maintenance and opeator error dominate risk. A disciplined, systematic and detailed analysis of systems functions, interfaces, and operating and failure modes is required to properly understand the relative importance of even the major contributors to postulated accidents. This is often referred to as Integrated Systems analysis. Complementary to its use in a Probabilistic Risk Assessment (PRA) for analyzing the probability of core damage and for studying plant modifications, the systems analysis is the key to addressing many of the important in-plant activities that assure safe and economical operation.

The enhancement of credibility of systems analysis methods and results has been the area of major emphasis of the EPRI research. This need was confirmed by a review [23] conducted during 1982 and 1983, of five large scale PRAs to provide a summary and an interpretation that would help technical specialists and management personnel understand the state of the art in risk assessment, the validity of the methods used, and the conclusions reached. The PRAs studied were Big Rock Point, Zion, Limerick, Grand Gulf (RSSMAP) and Arkansas Nuclear One-Unit 1 (IREP). The study found that the quantified description of human reliability, definition of common cause failures and degraded core analysis are areas that need strengthening for routine use. The introduction of systematic documentation and careful standardization of methods are also necessary if increased confidence is to be generated in PRA results. A further

TABLE 3. Seismic Center Program Elements & Schedules

General Assessment

- Quantification of Seismic Design Conservatism and Its Cost Impact (1984-88)
- Foreign Experimental Data and Analysis (1984-89)
- Interdisciplinary Seismic Technology and Engineering Analysis (1989)
- Inelastic Response Design Acceptance Criterion (1985-89)
- Seismic PRA Evaluation and Application (1989)
- Seismic Data Bank and Information Center (1988-89)

Earth Sciences

- Seismic Hazards and Seismological Research (Large Earthquakes in Eastern U.S.) (1985-89)
- Seismic Design Ground Motion for Nuclear Power Plants (1984-89)

Geotechnical Engineering

- Large-Scale Seismic Test (1984-89)
- STEALTH Applications (1984)
- SSI Test (1984)
- Improved Ground Motion Spectra (1986-89)
- In-Situ Soil Property Characterization (1985-89)
- Liquefaction and Soil Stability (1986-89)
- Simplified SSI Analylis (1987-89)

Structural Engineering

- Nonlinear Finite Element Code Development (1984-89)
- Piping and Fitting Dynamic Reliability (1984-87)
- Simplified Structural Design and Analysis Method (1985-89)
- Seismic Testing and Analysis (1984)
- Structural Response of Concrete Containment (1984-88)
- Seismic Mitigation (1984-87)
- National and International Participation of Large-Scale Structural Tests (1988-89)
- Simplified Design Handbook for Small Bore Piping (1985-89)
- Systematic Snubber Reduction (1985-89)
- Improved Floor Spectra Specification (1987-89)
- Improved Procedure of Combining Modal and Component Responses (1989)
- Building Piping and Equipment Coupling (1987-89)
- Concrete Damping (1989)
- Experience with Power Piping During and After Earthquakes (1984-87)

Equipment Qualification

- Equipment Qualification Program (Seismic Projects Only) (1984-89)
- Seismic Equipment Qualification and Capacity (1987-89)

study addressing sensitivity of the results to assumptions concerning Human Error, Dependent Events and Degraded Core Analysis has just been completed [24].

In parallel with several specific methods development projects, EPRI's Nuclear Safety Analysis Center (NSAC), in cooperation with Duke Power Company, has recently completed a full scale PRA of Duke's Oconee Unit 3 [25]. This study synthesized the best available methods for performing the various tasks of a PRA and documented in detail the methods, data, and assumptions for reference by future utility projects of the same type.

It is noteworthy that this study was able to draw a number of important engineering conclusions that were essentially independent of the assumptions and uncertainties in the analysis, and plant modifications were made on this basis. Two technical areas have yielded important results during 1984; (1) Common Cause Failure (CCF)[26,27] and (2) Systematic Human Action Reliability Procedures (SHARP)[28].

Common cause failures are generally considered to be events involving multiple hardware failures that have some common cause or origin. The causes are sometimes elusive and the events exhibit a perplexingly wide range of characteristics. Further, there is no consensus of exactly what constitutes a CCF event. There is thus disagreement about the frequency of CCF occurrence, accepted means of analysis, and systematic engineering practices for achieving defenses.

Initially, a peer survey and workshop was conducted to draw on the expert views of six nuclear plant reliability engineers with special experience in CCF issues. The peer survey emphasized the overriding need to develop a top-level classification system for equipment unavailability events, including their causes, before tackling the question of which are truly CCF. Such a classification system, requiring careful consideration of the proximate cause of component unavailability, has been devised. At a fundamental level it defines six mutually exclusive classes of events (Figure 5): independent failure, cascade failure, functional unavailability, conditionally independent failures, multiple cascade failure, and multiple functional unavailabilities. This scheme has been tested and extensively developed in one U.S. and one European benchmark experiment with the participation of thirteen organizations including six utilities and four national laboratories [26,27].

The impact of human error on plant safety has been a major concern since the President's Commission on Three Mile Island found that "the equipment was sufficiently good; except for human failures, the major accident at TMI would have been a minor incident". Techniques to analyze the reliability of human interactions with equipment have increased rapidly since then. However, such techniques need to be integrated in a consistent way with systems reliability analyses to produce a credible and usable probabilistic risk assessment. We have responded to this situation by developing a Systematic Human Action Reliability Procedure (SHARP)[28] for consistently and comprehensively incorporating human interactions into PRAs.

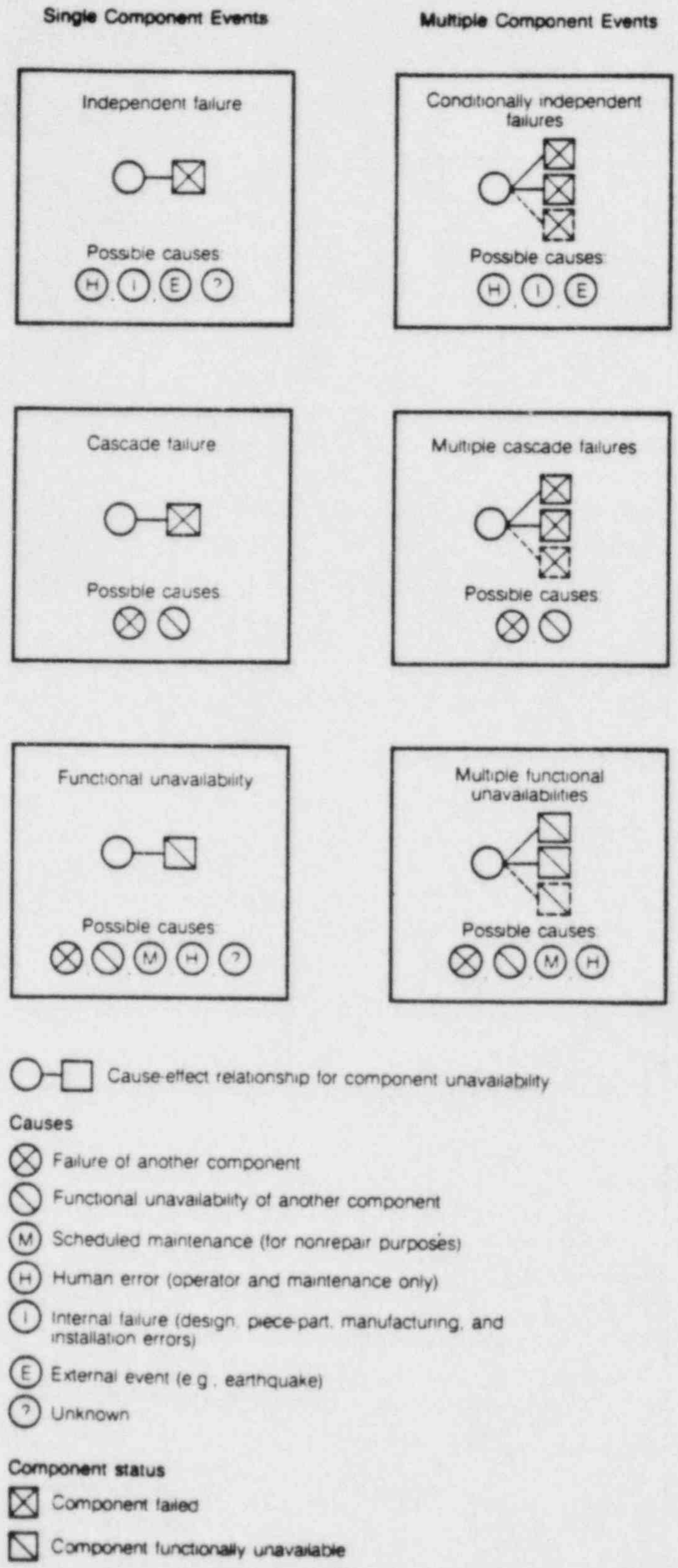


FIGURE 5. Component Unavailability Events Classification

Investigators reviewed selected PRAs to identify the implicit steps in these studies that analyze human interactions with equipment. These steps were modified according to suggestions made at an EPRI-sponsored workshop. Further modifications resulted from analyses of human interactions during different hypothetical accident sequences. Activities and rules were then added to help analysts implement the steps. The draft report was reviewed by independent reviewers, including international experts on human reliability analyses and PRAs and the IEEE Working Group on Human Performance.

The Systematic Human Action Reliability Procedure (SHARP) consists of seven specific steps that should be carried out by both human reliability analysts and systems analysts.

The SHARP steps are currently being embodied in a guidance document on Human Reliability Analysis by the IEEE[29]. They are also being subjected to a benchmark process during 1984 to further test their utility, and are already being incorporated into several PRAs.

SAFETY MARGINS DEMONSTRATION

Integral Systems Tests

With an increased emphasis on risk dominant small breaks, safety testing has given increased attention to small break flow phenomena and heat removal mechanisms. Operator actions assume greater importance, and the contribution of emergency procedures and operator decisions to both risk and accident progression require evaluation. As stated earlier, one of the objectives of safety research program is to produce large scale experimental data which supports the safety analysis of plants. The distinctive features of these plants (OTSGs, vent valves, and hot leg candy-cane) require experimental confirmation for small breaks and transients.

Within this joint NRC/EPRI/B&W Owners/Group Program, a raised- loop B&W design is represented by the OTIS facility - a scaled, one-loop, high pressure and full height facility; and a lowered loop design by the MIST facility, represent the two-loop primary system designs operating at full pressure. A series of fifteen tests were successfully performed in OTIS. The test facility is again scaled at full height and full pressure with a volume scaling ratio of 1/817. Over forty tests, in addition to shakedown and characterization tests, are planned in MIST for completion in 1986[30]. These will form a complete basis for code qualification.

In order to test and support the scaling rationale and interpretation of these large facilities, a test facility, built at SRI, also represents the primary system of a B&W plant[31]. The facility is, however, small in size (scale factor = 1296) and operates at lower pressure (7.8 bars/115 psia). The objective of this facility is to verify the scaling approaches suggested and to perform parametric studies on specific design aspects of the downcomer vent valves and candy-cane curvature[32]. The facility will be operational in spring 1985.

Separate effect studies refine our knowledge of key phenomena expected to take place during OTSG transients. These phenomena are phase-separation and liquid circulation in low-velocity two-phase natural circulation. Flow regimes are studied in large vertical pipes with pipe diameters varying from four inches to twelve inches, with entrance effects and candy-cane also modeled in order to study phase separation in that region. The first results have been compared to various flow regime maps (e.g., Taitel and Dukler) and the interruption of natural circulation has been determined for prototypical conditions (Figure 6). The twelve inch pipe study will determine potential variations of flow regimes due to pipe diameter. These studies show, in a preliminary manner, that the circulation and flow regimes are predictable, and are not vastly different from prior work(33,34).

Because of flow stratification, Small Break Critical flow is being studied in a joint project with NRC. Measurements are made of the break flow rate in a horizontal pipe in the presence of stratified two-phase flow. The critical flow out of the break, under these circumstances, is a strong function of the circumferential location of the break. The experimental data will be compared against an analytical model for Small Break Critical Flow being developed by Hardy and Richter(35).

Pressurized Thermal Shock

At low flows under natural circulation, the likelihood of thermal shock to the RPV had been a concern due to cold water injection. Criteria have now been established for RPV integrity, and the EPRI R&D is in a completion phase. Transient thermal mixing experiments have been performed in the 1/2 scale test facility (cosponsored by EPRI and NRC) located at Creare, New Hampshire(36).

To interpret these and other data, mixing analyses have been conducted, using both a multi-dimensional thermal hydraulics computer code (COMMIX-1A), and simpler methods based on physical modeling. Both have been very successful. They are supported by basic experiments at UCLA on heat transfer in vertical annuli and gaps. It has been conclusively demonstrated that the COMMIX-1A computer code with mass-flow-weighted skew-upwind scheme, proper geometric modeling and an improved turbulence model can be used to analyze PTS mixing(37-41).

The simpler models are based on five segmented volume model for transient cooldown analysis in a reactor cold leg and downcomer under stagnant loop flow utilizes mixing correlations based on various test data. This simple model gave reasonably satisfactory predictions for transient cooldown with dramatically reduced computational time(42,43).

Reactor Coolant Pump Performance

In small breaks and transients, the flow performance and head degradation of the pump affect the course of the transient. Hence, the adoption of pump trip criteria. An analytical model has been

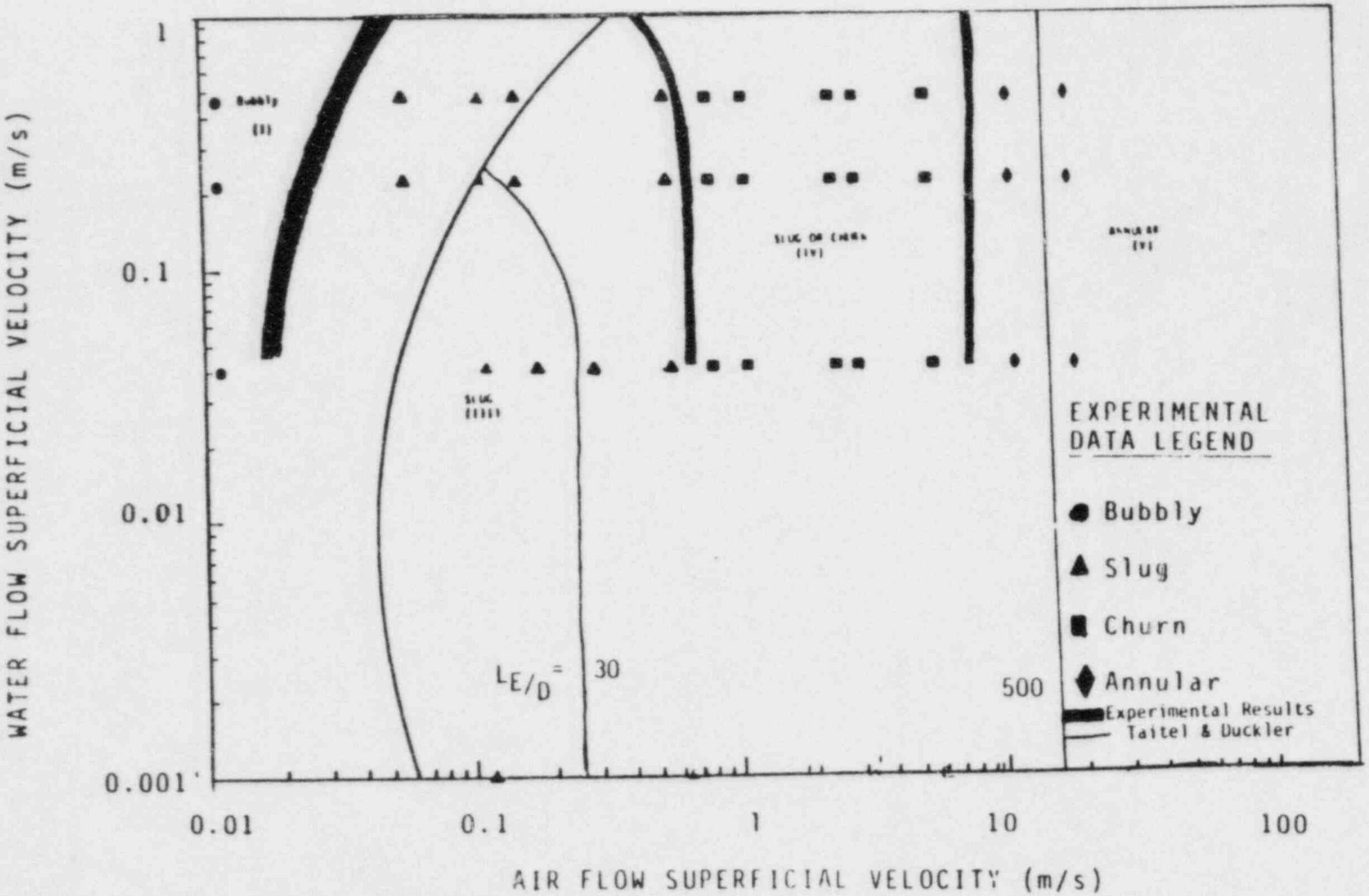


FIGURE 6. Comparison of experimentally observed flow regimes in the vertical section of the hot leg with Taitel and Duckler analytical results.

developed that predicts the reactor coolant pump performance under two-phase flow conditions[44]. Unlike other existing models that rely solely on empirical correlations and treat the pump as a passive black box, the new analytical model is based on mechanistic principles of two-phase flow dynamics with the pump. The model predictions agree favorably with existing test data for both air/water and steam/water systems.

Activity Transport and Steam Generator Tube Rupture (SGTR)

Steam Generator Tube Rupture (SGTR) is treated as a generic safety issue related to the Pressurized Water Reactors (PWRs). A question related to the issue is how to evaluate consequences of radioactivity release following a postulated SGTR event. Related considerations pertain to leak detection, leak duration before break, operator action to minimize the activity release, single versus multiple tube ruptures and the consequence and accident progression with other occurrences (such as stuck open PORV along with SGTR).

By using codes such as RETRAN02, RELAP4 and MMS-02, it is possible to evaluate the PWR plants, and the Semiscale and MB-2 experiments. Based on the RELAP4 and RETRAN02 results, a small sensitivity of the peak clad temperature to a single and multiple SGTR is observed. The MMS-02 code has been used to evaluate recent Semiscale SGTR test series representing one, five and ten tube ruptures, as well as the Prairie Island SGTR event, the results of which are presented in Figure 7[45-47]. The Duke Power Company is also using the MMS code, along with other codes, to verify the W EPGs during the SGTR event[48].

What these analyses demonstrate is the detailed modeling of SGTR sequences are possible and predictable. Emphasis is now turning to the further validation of activity transport methods.

Large Scale Nuclear Tests

EPRI participates in the OECD sponsored international LOFT Consortium. The successful completion of thermal hydraulic tests on small and large breaks in 1983 and 1984 has paved the way for two important activity release tests. There FP-1 and FP-2 are designed to characterize the release of fission products from the fuel (volitile, noble and aerosols) and to determine the transport and retention of activity in the primary system.

Multi-Dimensional Effects in Core Cooling

Design basis safety analyses have relied on simplified, one-dimensional core assessments and have therefore placed a limitation of peak-power levels in reactor cores. What was unknown was whether such one-dimensional assessments were adequate for large PWR cores or whether a model based on three-dimensional flow and heat transfer effects would provide a larger or smaller safety margin.

A 1700 pin, scaled reactor core model was built and measurements were taken of the three-dimensional flow and heat transfer that

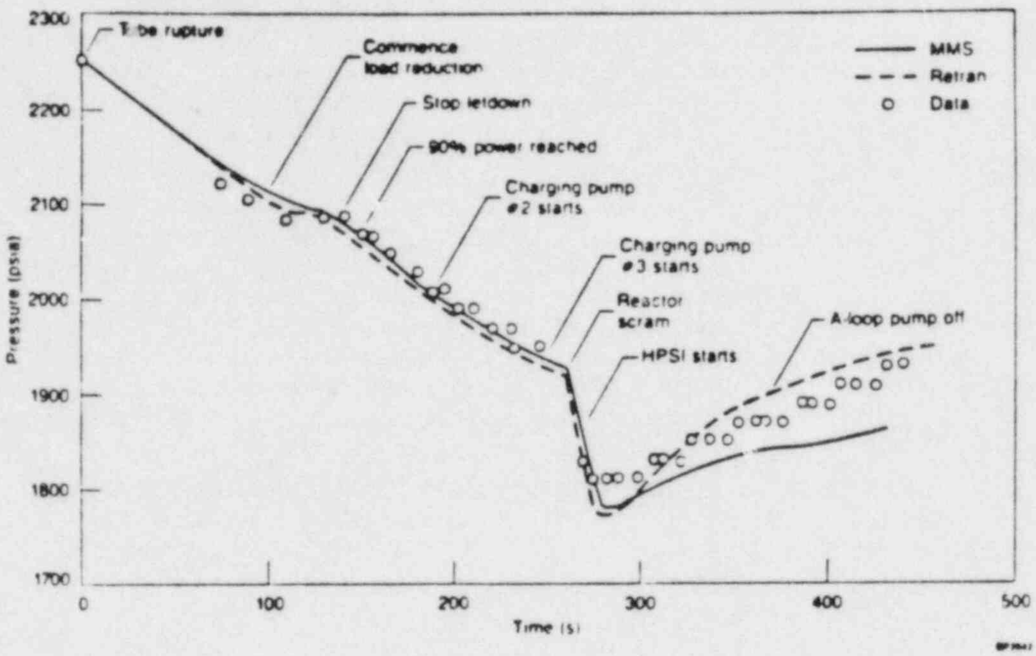


FIGURE 7a. Praire Island RCS Pressure Versus Time

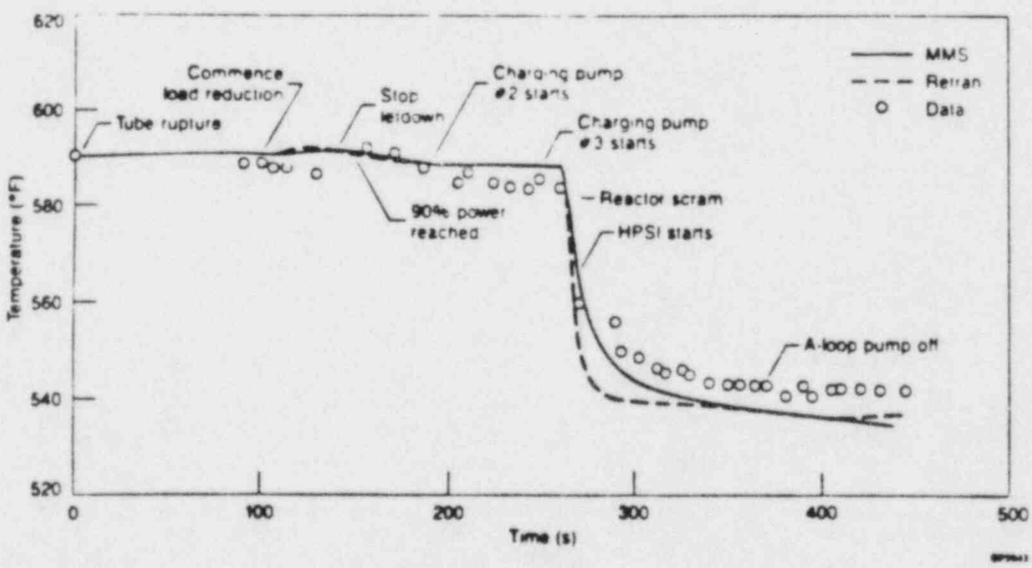


FIGURE 7b. Praire Island Hot Leg Temperature Versus Time (A-loop)

occurred during core reflooding. These small-scale results agreed with full-scale reflooding studies, confirming scaling and validity of the results. The analysis supports the use of multiple parallel channels to determine core-wide flow phenomena[49].

OPERATIONAL AND SAFETY IMPLEMENTATION OF DIGITAL TECHNOLOGY

As the computer revolution proceeds, the nuclear industry is implementing enhanced computer and digital techniques for safety and operational enhancements. Results of prior research in Disturbance Analysis Systems provided the technology base, and the post-TMI SPDS requirements the impetus for change in signal presentation for operator support, actions and aids[50,51]. Substantial upgrading of computer hardware in plants allows more sophisticated applications to be developed for operator support. Greater attention is being given to on-line validation of input signals for computer applications whereby the signals are validated before display[52].

The integration of displays with operating procedures enables superior coupling between problem detection and its resolution[53]. Trends beyond the next few years will move towards more intelligent software[54]. Artificial intelligence technology may play a pivotal role in future applications[55].

Signal Validation: Application and Promise

The reliability of input data is of increasing concern to utilities as the sophistication of computerized operator support functions becomes greater. The masking of low level data errors or sensor deterioration with significant consequences must be avoided. There are studies now underway to implement validation of BWR and PWR parameters.

The design of validation for BWR suppression pool parameters has been completed[52]. An example of signal validation architecture is shown in Figure 8. The design accommodates current and upgraded suppression pool instrumentation, validates safety relief valve positions and provides a single validated measurement for the suppression pool bulk temperature. The description of analytic measurements is given in Table 4. Another BWR parameter warranting signal validation is the reactor vessel water level. This is desirable to reduce one cause of plant trips. The signal validation of the design is in progress[56].

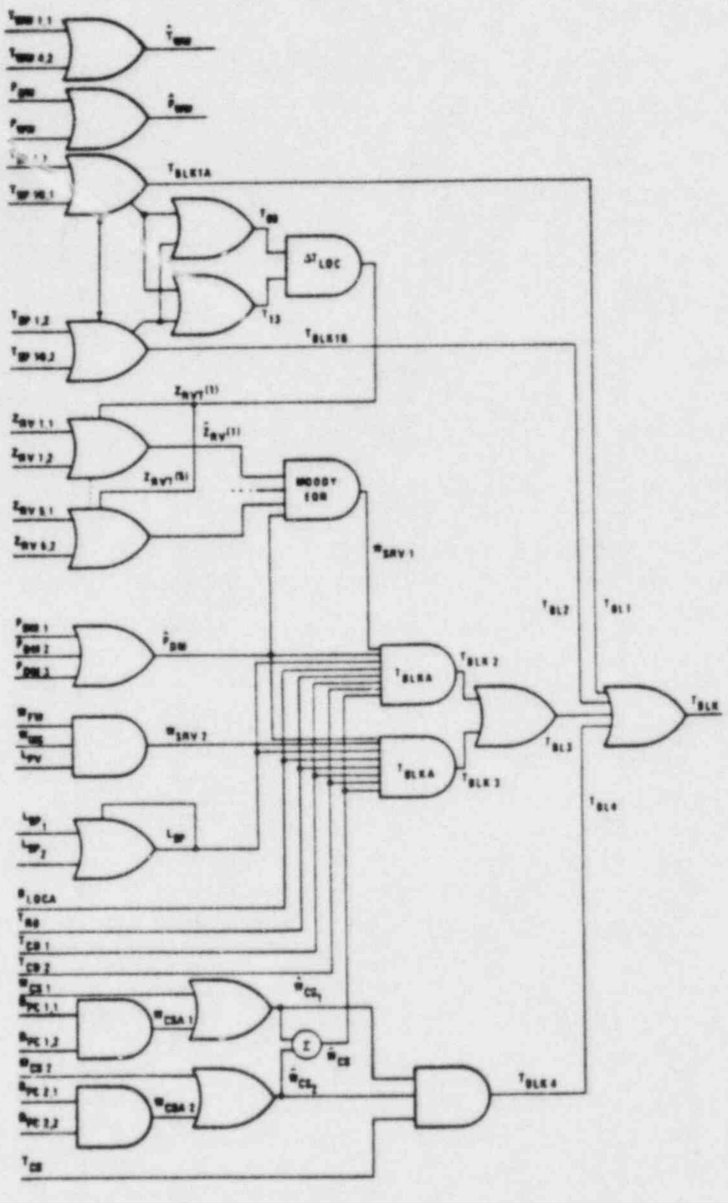
The validation requirements of the PWR variables have been identified by a consensus of utility participants, as necessary inputs to the PWR SPDS critical safety function algorithms. The host plant for the demonstration is the Northeast Utilities Millstone-2 Nuclear Station[57,58].

On-Line Core Monitoring Capability

In parallel with computerized support for plant emergency operations is the development and use of advanced process management functions

Suppression Pool

Instrumentation Readings



T_{BLK} Validation

FIGURE 8. Top Level Signal Validation Flow Diagram (Reference 52)

Legend

Variables	Subscripts	
T-Temperature	WW-Wet Well	MS-Mainstream
P-Pressure	DW-Dry Well	PV-Pressure Vessel
L-Level	SP-Suppression Pool	RB-Reactor Building
F-Flow	RV-Relief Valve	CD-Containment Spray Discharger
D-Discrete Signal	DM-Dome	CS-Containment Spray
	FW-Feed Water	PC-Containment Spray Pump

TABLE 4. Analytic Redundancy Measurements

ZRVT	SRV position from temperature rise in quencher bays
WSRV1	Total SRV steam flow, using Moody model
WSRV2	Total SRV steam flow from change in steam/feed-flow mismatch
WCSA	Containment spray system flow from pump discrete (on/off) signals
TBLK2	Bulk temperature of suppression pool using mass/energy balance, with the steam input flow from WSRV1
TBLK1A	Suppression pool bulk temperature from a volume-weighted average of division A pool temperature sensors
TBLK1B	Suppression pool bulk temperature from a volume-weighted average of division B pool temperature sensors
TBLK4	Bulk temperature, equal to the containment spray system suction temperature when the system operates

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TBLK1B	Suppression pool bulk temperature from a volume-weighted average of division B pool temperature sensors
TBLK4	Bulk temperature, equal to the containment spray system suction temperature when the system operates

for normal operations: on-line software for core monitoring and power distribution control. These core monitoring systems couple plant data streams to multi-dimensional nodal analysis routines and provide power distribution and prediction, exposure core limits, trends, anamoly detection and control rod position searches, etc. By careful benchmarking against plant data (e.g., for Salem, ANO and Summer), accuracy of the prediction of core rating and power distribution can be obtained[60]. A new development is the development of a software framework for implementing various core monitoring systems called ACMF[61].

Digital Applications Controls in Power Plants

The primary control functions in LWRs are performed by a small number of control loops, which use analog technology. The feedwater control system, which is one of the loops, is one of the largest contributors to plant outages in both BWRs and PWRs. Recent developments in high-reliability microprocessors and applications of fault-tolerant microprocessor computer systems in the nuclear industry have prompted conceptual design studies at EPRI to evaluate Digital design of the feedwater control system for PWRs and BWRs[62-64]. Implementation of these designs at host plants is anticipated in 1985 and 1986.

CORE PERFORMANCE LIMITS AND PLANT TRANSIENT EVALUATION

Automated and Standardized Physics Methods

The Advanced Recycle Methodology Program (ARMP) calculational sequence for PWRs has been recognized as requiring excessive manual transfer of information between codes. As a result, a highly automated ARMP Version is in preparation. The standard code flow sequence (Figure 9) has required independent calculations for Gadolinium, burnable poison, and control rods. The new procedure (Figure 10) has replaced these with software heavily benchmarked against data. The resulting package has a built-in default option which permits a straightforward calculation through the fuel cycle including all branch calculations needed for feedback analysis. The MOD-00 calculational sequence stops short of including NORGE-P, hence, does not yet fully eliminate all manual information transfer. The MOD-01 version will include NORGE-P and provide all input needed for nodal analysis (NODE-P2 or SIMULATE-E)[65,66]. The method is expected to result in significant savings in engineering manhours.

Core Thermal Methods

An advanced fully compressible six equation 3D space-time version with full vessel modeling capability (VIPRE-02), which can also perform in the VIPRE-01 mode, will allow non- symmetric flow conditions to be analyzed[68-70]. The power distribution is an input variable, but because of the difficulty of developing 3D space-time power distributions for use with rapid transient thermal hydraulic calculations, an advanced neutronic code with rapid transient

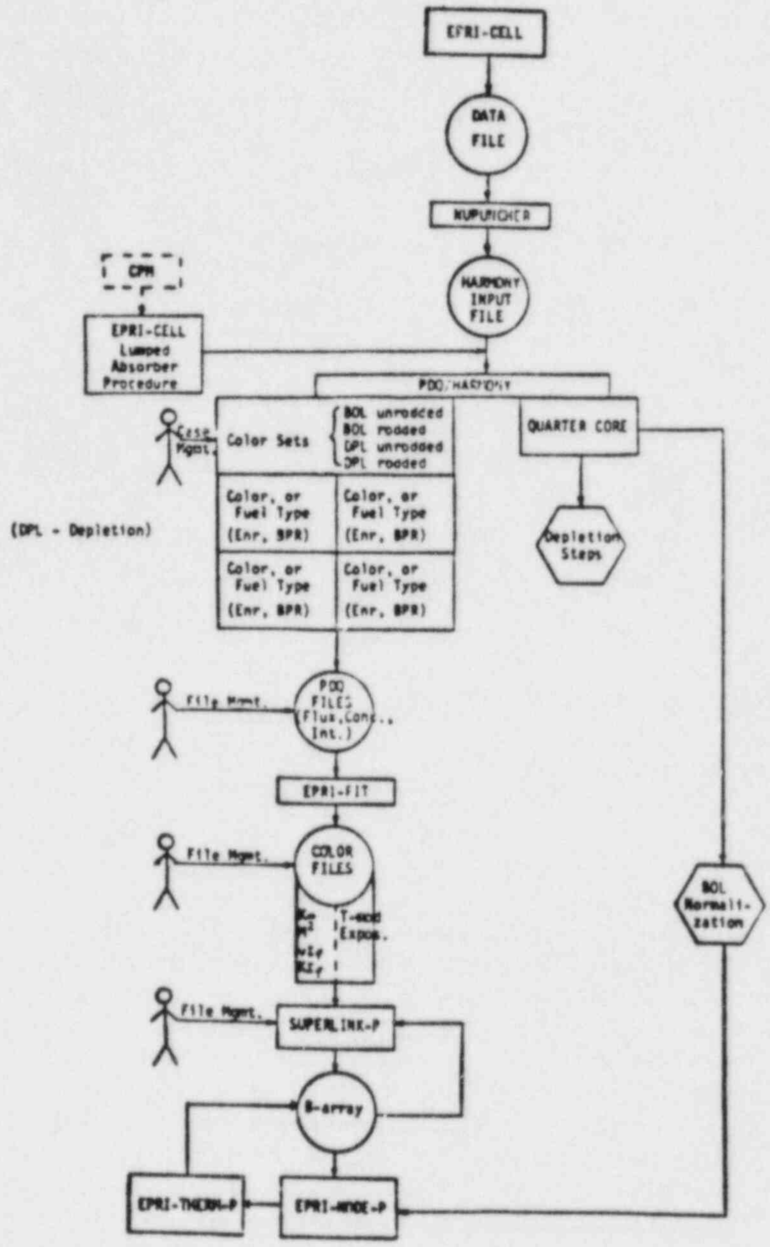


FIGURE 9. PWR Calculation Flow for ARMP

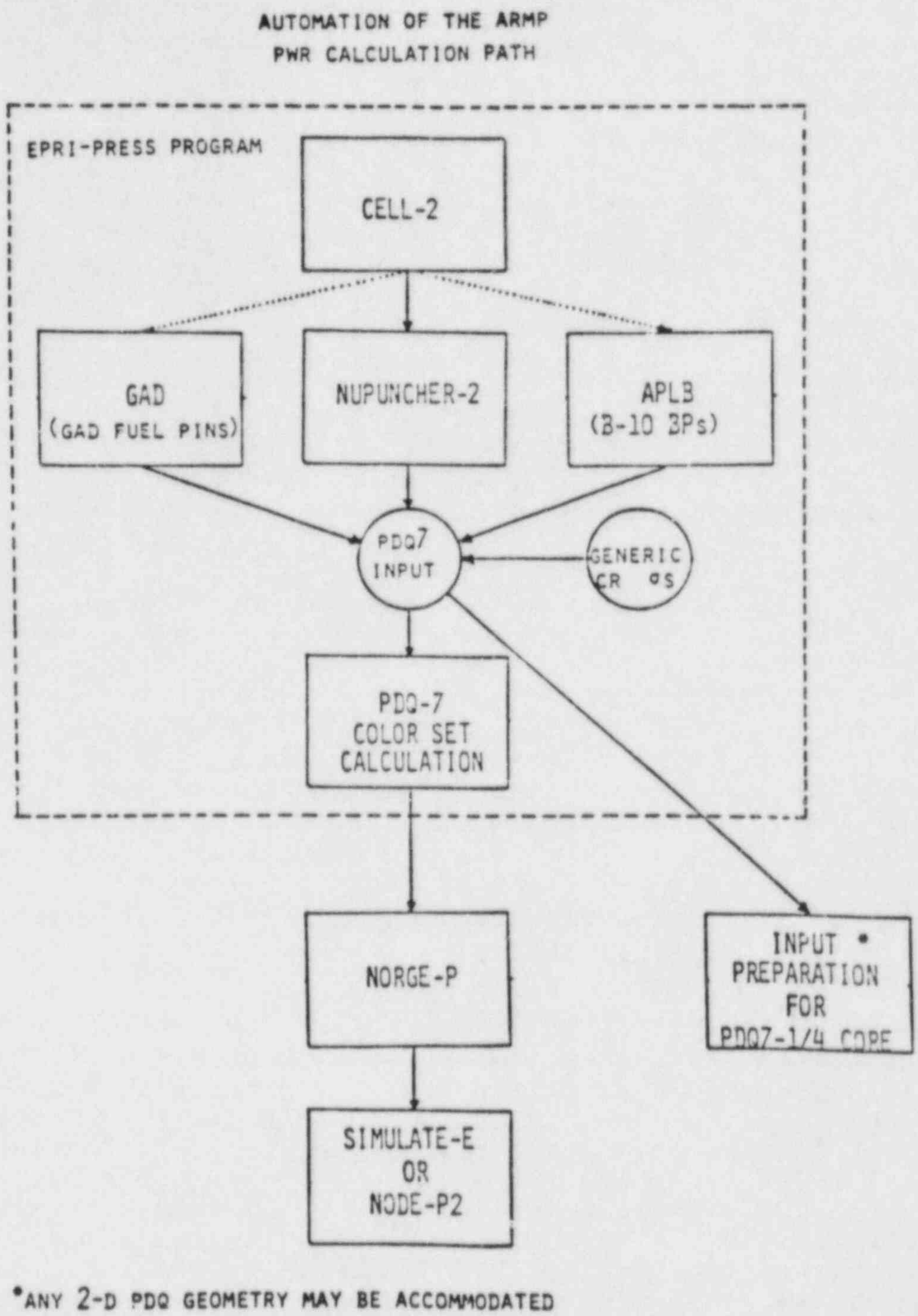


FIGURE 10. Automation of the ARMP PWR Calculation Path

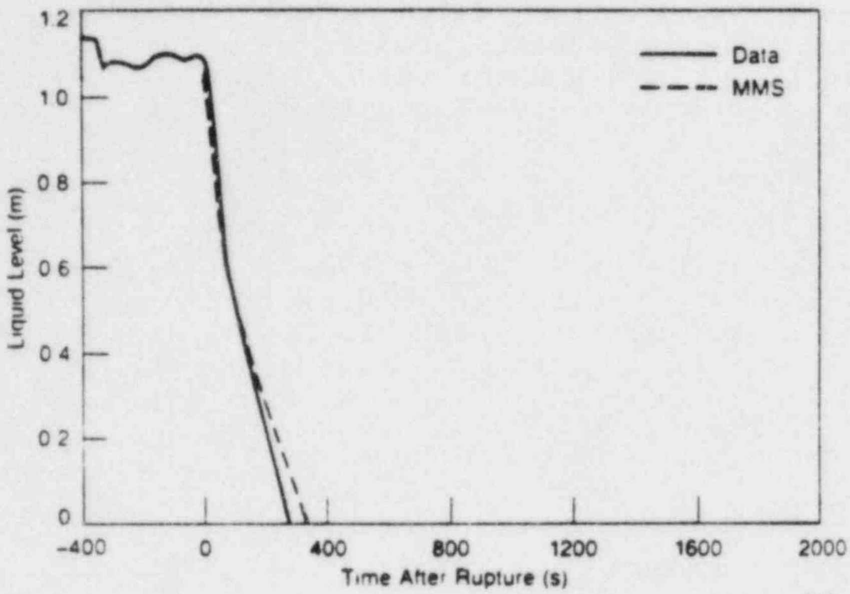
capabilities is being developed[71]. The ARROTTA (Advanced Rapid Reactor Operational Transient Analyzer) code is based on QUANDRY methodology and efficient implicit programming techniques[72]. A full two group capability including explicit baffle/reflector treatment with six delayed groups provides a state-of-the-art tool for 1/4 to full core transient analysis. This code starts from base cross sections and feedback coefficients from the NODE-P2[73] or SIMULATE nodal codes at any time during a cycle and accepts them from the existing nodal code restart files. The base code ARROTTA-01 contains the five equation hydraulic models. In early 1985 VIPRE-02 and ARROTTA-01 will be interfaced to produce a consistent full vessel model capable of modeling non-symmetric rapid or long term xenon drive, transients.

The implementation of the ARMP computational package, under a full reload methodology (RASP), has been underway. A preliminary code version has now been released to interested utilities[74].

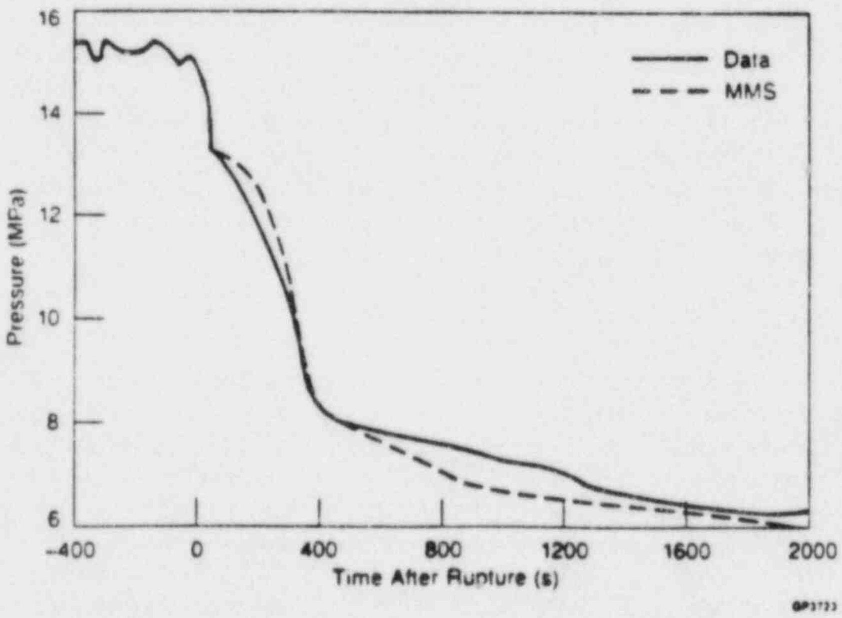
Transient and Operational Analysis: Modular Modeling System (MMS) Code

The operational transient analysis version, MMS-01, for the pressurized water reactors and the fossil plants was released in 1983 after undergoing extensive validation by eleven pre- release users group members. Verification of the scoping tool for the accident analysis of both BWRs and PWRs, MMS-02, was performed complemented by independent evaluation of the code by six to eight utilities[75]. Among the accidents and transients used for code validation were: (1) the Peach Bottom tests, (2) the TMI-2 accident, (3) TMI-2 overcooling transient, (4) the LOFT L3-7 small break transient, (5) Prairie Island steam generator tube rupture event, and (6) Semiscale SGTR test comparisons[76-81]. Some sample LOFT L3-7 results are presented in Figure 11[75].

Further comparisons with RETRAN-02 code (as a benchmark) predictions are also performed for steamline and small breaks for PWRs[75]. The utility industry is using the code for a variety of applications, such as control systems evaluations in switching to digital systems, BOP and whole plant modeling, and procedures evaluation. The largest model developed to date is by Middle South Services, consisting of 208 equation model for the Grand Gulf nuclear plant with a very detailed balance-of-plant linked to the BWR module and the major control systems for the turbine inlet pressure, feedwater, recirculation pump speed and the reactivity. The simplest model was developed by the Duke Power Company with a thirteen state MMS model of the feedwater heater drain system. This small model exercise indicated that consideration of key design information (a five second time delay associated with the dump valve) could be vital in order to accurately predict system transients like pump trips[75]. Another example of the utility application was a situation where a "PID" controller was replaced by a simple proportional controller[75].



a. Pressurizer Level



b. Primary System Pressure

FIGURE 11a,b. MMS Calculations Compared to LOFT L3-7 Data (Small Break LOCA)

Transient Plant Data

As is evident, adequate plant transient model validation requires reliable plant transient data! There are several plant transients and start-up data that are either available or will become available. For instance, EPRI is gathering start-up test data on Southern California Edison's San Onofre-2 (CE plant) and Middle South Services' Grand Gulf (BWR-6). The data bank inclusions can be found in Reference 80.

Simulator Qualification Methodology

It is necessary to train operators on simulators. By integrating the training requirements with the expected plant states, a means has been developed to systematically qualify simulators. This does not require other than a careful evaluation process and analysis . This procedure, developed in association with a utility advisory group, is reported[83,84].

CONCLUDING REMARKS

This morning we have tried to give you an overview of EPRI's work in the area of reactor safety. We have tried to avoid going into too much technical detail -if you really want to know the details on individual activities, the references are generally available. We have also tried to let you know that we are trying to structure the research so that results are self evident or can be described in language that can be understood by non-specialists.

Overall, the current research conducted by EPRI has been both stimulating and encouraging. The next few years will produce conclusive results on the Source Term, Seismic Requirements and Systems Analysis. All these results will be heavily influenced by large scale and visible demonstrations to support continuing transparency of analysis. We also will have the opportunity to implement and test digital technology in operating LWRs. On-line core monitoring and signal validation efforts will help in evaluation of standards for on-line software.

Despite all good news about our research results, we must be aware of one very important fact. We must learn to effectively communicate the results of our research. It is easy to become so impressed with the reams of computer paper and stacks of reports we generate each year that we forget to answer the questions that the utilties, their ratepayers and their regulators are asking.

As we close, we would like to refer once more to the opening analogy between the airline and nuclear power industries. If you recall, one of the major differences in the two industries is their methods of communication. While the airline industry has tried to be personable and friendly, we have tried to be accurate and scientific: while the airline industry has created images of families being reunited at Christmas, we have created images of bar charts and concrete containment buildings and cooling towers. The airline industry operates in harmony with the public. We will close with the question: What can we as researchers do to contribute to that harmony for our industry? Remember, a truth that cannot be understood can become or be perceived as an error!

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APPENDIX A

Aircraft vs. Nuclear Industry

		Aircraft Industry	Nuclear Industry
1.	Mission:	To provide continuous, safe and rapid transportation service to public	To provide continuous, safe and diversified power to public
2.	Major Vendors:	Aircraft Manufacturer: Boeing, McDonnel Douglas & Lockheed Engine Suppliers: GE, P&W & Rolls Royce (Sub-Contractors to Aircraft Manufacturer)	NSSS: GE, W, CE & B&W BOP: Six to twelve A/ES
3.	Customers:	Domestic & Foreign Airlines	Mostly U.S. Utilities
4.	Regulations:		
	(a) Safety Monitoring and Regulations	FAA: Chartered to promote Civil Aviation Industry and to Regulate Safety	NRC: Chartered only to Regulate the Industry Safety
		NTSB: Accident Investigations without Regulatory Power	INPO: Investigations into the Cause of Abnormal Events
	(b) Economic Regulations	CAB: Issued Route Awards and Tariff Approvals until Recent De-regulation	PUC: Allow Rate Adjustments
5.	Licenses Required:	Type Certificate by Manufacturer; Operating Permit by Airline	Construction License by Utility; Operation License by Utility
6.	Analysis Requirements		
	(a) Design Analysis	Aircraft Manufacturer/ FAA Designee/FAA	Vendor and A/ES
	(b) Design/ Manufacturing/ Construction Integration	Aircraft Manufacturer	Utility

Aircraft vs. Nuclear Industry (cont.)

(c) Maintenance Programs	Airlines/FAA	Utility
(d) Regulatory Induced Re-Design/Verification Analysis	Airworthiness Directives	Continuing Task for the Utility to Meet Regulatory Requirements
7. Analytical Methods and Testing		
(a) Structural Analysis	Sophisticated Finite Element and Stress Analysis Codes	Technology Same as Aircraft Industry
(b) Simulation Codes	Aircraft Manufacturers use Fluid Mechanics and Heat Transfer Codes. Engine Performance Codes are used by the Engine Manufacturers.	NSSS Vendors use Sophisticated Technology for Design & Verification. A/ES use Heat Balance Codes. Simulation Codes Developed by NSSS, EPRI, and NRC are Available for overall System Simulation.
(c) System Testing	System Simulators Integral Part of Design Process	Training Simulators Increasingly Used in Re-Design Process of Control Rooms in Some Plants
	Model Specific Full Scale Replicas of Structure, Wind Tunnels and Development Cabs, are Used	Scaled Testing of Components and Experimental Verification of Phenomena are Used for Generic Plants.
	Component Static Tests are Conducted; Full-scale Prototype Flight Tests are Performed	Accident Simulations in Experimental Reactors, and in Scaled-down Test Facilities are Conducted Plant Start-up Tests Performed

Aircraft vs. Nuclear Industry (cont.)

	(d) Reliability Analysis	Manufacturer's Reliability Data Base; Extensive Traditional Reliability Analysis Throughout Design Process. Reliability Monitored in Operation by Airlines	Simple Failure Criteria plus PRA (Recently Introduced in Operating Plants and Plants under Construction)
8.	On-Line and Automated Systems:	Extensively Implemented; Design Requirements in Many Cases (Developed Through Operation)	Very Limited Implementation; Not a Design Requirement. Increasing Emphasis as Retrofit Design Improvements in Some Plants.
9.	Operational Training:	Type Specific Simulators Training Mandatory; Six Month Requalification Requirement for Airline Pilots "Similar Aircraft" Training not Acceptable	Plant Specific Training Requirements Recently Introduced. Increasing Emphasis on First Principles Based Simulator Usage in Plants. "Similar Plant" Experience Acceptable
	Maintenance Training	Explicit Job Level Training Task Qualification	Redesigned Training Programs After TMI-2 Accident
10.	Operators:	Generally College Educated, Well Paid and Motivated; Jobs Perceived to be Glamourous	Generally, High School Educated; Jobs Perceived to be Challenging
	Maintainers	Technically Well Trained	Technically Well Trained
11.	Public Perception:	"<u>Very</u> Favorable" now (Risk Perceived to be Minimal and from Personal Choice; Industry Fulfills a personel Need). Mode of Transportation Considered "Acceptable"	"<u>Very</u> Unfavorable" (Risk Perceived to be Potentially Devastating and Imposed Upon; Industry not fulfilling a Personal Need). Mode of Electricity Generation Considered "Risky"

Aircraft vs. Nuclear Industry (cont.)

12.	Research Funds and Results:	DOD: Won Wars for U.S.A.	DOD: Created 'High Level' Waste and Nuclear Bombs
		NASA: Promoted New Design Concepts for Aerodynamics, Power Plant & Structure**	DOE: Contributed to Basic Designs
		FAA: Operational Problems and Advanced Technology Implementation	NRC: Research on Safety Problems
		Vendor R&D: Resulted in Commercial Products	Vendors: Commercial Products
			EPRI: Utility Support
		Customer R&D: Training Methods, Operation Analysis & Maintenance Technology (Repair Techniques)	Customer R&D: Some Major Utilities Perform Their Own R&D; EPRI & INPO are Supported
13.	Marketing	Aggressive and Personable	Muted Cost-Effective
14.	Media Reporting of Incidences	Major Crashes Make Headlines; Rapidly Forgotten	Minor Instances Make Headlines; Nagging Questions Prevail with "Residual" Risk
15.	Product Warranties	Extensive Coverage/ Reliability Oriented/ Logistic Support Until Reliability Meets Goal	Limited Coverage on NSSS, Safety and Availability Oriented, Support from NSSS and BOP Vendors.

** Airline industry has successfully resisted NASA attempts to design entire transport aircraft.

PIPING RESPONSE TESTING ASSOCIATED WITH PIPE RUPTURE

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1. Introduction

Appendix A to 10 CFR, Part 50 requires that systems and components important to safety "be appropriately protected against dynamic effects, including the effects of pipe whip and discharging fluid (jet impingement), that may result from postulated piping ruptures." Reactor designers and operators can provide this protection in many ways, from initially designed separation to avoid direct impact to retrofit devices, such as pipe whip restraints. The assumptions used in evaluating pipe whip and jet impingement greatly influence the type of protective measures taken. A typical current-generation pressurized water reactor (PWR) can have 250-400 pipe whip restraints, which can cost an estimated $20-$40 million per unit to design, procure, and construct[1]*. This cost estimate does not include additional operating costs associated with in-service inspection and maintenance that results from difficult access and other design problems. Even more important, the difficult access increases occupational radiation doses for maintenance personnel.

The size and number of protective devices is a direct result of the assumptions and procedures used in design. Generally, industry practice has been to use conservative engineering judgment and logic for ensuring upper-bound considerations. To improve current practice, the industry needs criteria and methods based on mechanistic assessment of actual pipe rupture phenomena and its effects, including studies of pipe whip impact, jet impingement, and pipe rupture and depressurization. EPRI has sponsored both analytic and experimental research in these areas to meet the nuclear industry's needs.

In this paper, the most recent test results on piping response associated with pipe rupture, namely pipe whip impact and pipe rupture and depressurization, are reported. The tests performed do not represent any prototypical design configuration. The main objective was to generate a data base for validating the highly-nonlinear methodology required for realistic pipe rupture induced response evaluation and to quantify the conservatism in current simplified industry practice in analyzing pipe rupture related situations.

2. Pipe Whip Impact

France's CEA and FRAMATOME (with major funding from EPRI) joined EPRI in the pipe whip experimental effort at CEA's Cadarache Laboratory [2]. The major objective is to study the forces applied by the pipe to a given target during the pipe whip situation. It is intended that with the data collected conservative assumptions associated with pipe whip can be evaluated and realistic analysis models and designs can be qualified.

*Numbers in brackets designated references listed at the end of paper.

2.1 Test Setup

The AQUITAINE II facility operated by CEA at Cadarache was utilized to perform the experiments. The facility was designed for operating pressure at 2494 psi and temperature at 644°. The main components of the facility, Figure 1, consist of a steam supply vessel with a capacity of 8.83 ft^3 heated by 12 electrical heaters in the lower part of the vessel, a pressure raising pump connected to a tank fed with demineralized water, and a circulating pump for homogenizing temperatures in the vessel and the test section. To record the initial conditions and transient system response histories during the test, extensive instrumentations were installed. Figure 2 shows the transducer arrangement on the vessel for pressure and temperature measurements, and Figure 3 shows the load cell (E1 to E5) arrangement for measuring the vessel reaction response.

2.2 Test Specimen Design

Figure 4 shows the configuration of the test specimen which is either of three-inch Schedule 80 or three-inch Schedule 10 pipe. Test pipes are made of ferritic steel TU42C (French reference) which is equivalent to A106 Grade B. Instrumentations are installed on the test specimen to record temperature and pressure histories before and during the test.

Pipe ruptures were initiated by the detonation of an explosive cord wrapped circumferentially at a given level along the vertical section of the test specimen to induce an instantaneous circumferential break (less than one millisecond). At the location that the explosive cord was installed, the thickness of the pipe was machined down to yield a remaining thickness of 0.14 inch. With the steel vaporized by the explosive cord, instantaneous pipe rupture with minimum deformation at the breach location was achieved.

Two different types of impact targets were designed. One was a rigid type target (Type I) to measure directly the impact force and the second type (Type II) was of concrete slabs to study the performance of the slab and the interaction between the concrete and the impacting pipe. Both types of targets were equipped with load cells for impact force measurement.

2.3 Test Matrix

A total of 16 tests were considered. Eleven were Type I targets and the rest were of Type II. Tables 1 and 2 show the entire test matrix. The testing parameters are denoted in Figure 5 and 6. For Type II tests, all five tests have the same geometric parameters as follows:

Pipe : 3 inch Schedule 80
L_1 : 10.5 feet
L_2 : 6 feet
δ : 5 feet
R : 4.5 inches

The concrete slab target parameters are given in Table 3.

The test matrix was designed so that data covering a wide range of parametric variations can be collected for more indepth understanding of the impact phenomena and qualification of analysis models and methodologies.

2.4 Test Results

The test results can be discussed in two aspects. One is the whip phase which considers the pipe motion from the time of break to the first impact on surrounding structures or targets. The other is the impact phase which considers the direct impact between the pipe and the target.

The whipping phase characteristics depend on a number of physical parameters such as dimension of the pipe, strain rate, blowdown force intensity, displacement and deformation ranges, and mass of the terminal straight broken end. One of the most important measurements during this phase is the time elapsed at the time of pipe break to the first impact of pipe on target. The longer the whipping duration is, the greater the kinetic energy and, correspondingly, the impact force. Table 4 summarizes the principal results obtained during the whip phase. One finds, in general, that the pipe whip duration is longer whenever the gap is greater. However, the length of the vertical part has a great influence on pipe whip duration (37ms for Test 3 and 7.7ms for Test 10) since this vertical pipe controls the inertia of the moving test section and has significant influence on whipping duration. For Type II tests, one notes that Tests 2, 3 and 5 have consistent results. The longer whipping duration of Test 1 was probably due to the obstruction of the anti-smoke structure (block smoke caused by explosion for better visibility of test section). For Test 4, the initial cooler temperature in the test section led to a smaller jet thrust than the other cases and, consequently, the longer whipping duration.

The impact phase results are summarized in Tables 5 to 7. In this phase, the main measurement is the impact force which is the force exerted on the target (or slab) by the pipe during the contact. The other measurements which provide information to interpret the impact results are impact duration and residual pipe crushing. These two measurements are interrelated in that the larger the pipe crushing the longer the impact duration which in general leads to smaller impact force. The concrete slab results show that depending on the slab strength and thickness, the pipe may penetrate the slab or induce hairline cracks in the back face only. The first test showed very small effect on the slab, and the second one showed total penetration of the pipe into the slab. For Tests 3 to 5, different degrees of pipe penetration into the slab were observed but all were stopped by rebars. Because of the way the slab was supported and the slab stiffness due to varying rebar ratio and thickness, the direct impact force between pipe and the slab can only be inferred through analytical modeling of the test.

Analytical correlation of selected Type I tests was performed [3] using the ABAQUS-EPGEN finite element code [4] and the simplified approach recommended in ANSI 58.2 [5]. Comparison of measured impact loads to calculated ones show that ABAQUS-EPGEN results agree well with measurements on the upper bound, and the ANSI approach predicts loads three to four times the measured ones.

3. Pipe Rupture and Depressurization

The design acceptance requirement, which assumes an instantaneous guillotine pipe break, has resulted in very conservative pipe restraint systems and pipe whip barrier designs. The industry needs quantitative results backed by experimental data to define more-realistic initial conditions in pipe whip design. Recognizing this need, EPRI sponsored a test program of high-energy pipe leak and break experiments [6]. Wyle Laboratories was contracted to perform the study.

3.1 Test Setup

A high-energy flow facility was constructed to perform the pipe leak and rupture experiments under LWR conditions up to and capable of 2400 psi. As shown in Figure 7, the major facility components were a 200 cubic foot steam supply vessel, and two 12-inch diameter feed lines connecting to the test specimen. A boiler and a nitrogen tank were connected to the steam supply vessel to provide specified LWR temperature and pressur conditions. All steam supply lines and the vessel itself were fully insulated to reduce heat loss. Extensive instrumentations were installed as shown in Figure 8 to record the initial con ditions and transient system response histories during the test. Pressure and temperature histories at various locations provide thermalhydraulic conditions of the blowdown during and after the crack initiation. Strain gages in the vacinity of the crack, and the two load cells at the supports of the test specimen furnished crack deformation and jet thrust reaction force information.

3.2 Test Specimen Design

The first phase of the program concentrated on the axial crack configuration only. The test specimen was designed to have flaw length that was either subcritical or supercritical with the critical flaw length determined using the net section collapse criterion derived in [7]. Since temperature and pressure of a specified LWR system condition should occur concurrently with the crack opening to satisfy the program objective, special considerations of flaw design were required.

After some evaluation and experimentation, it was found that the explosive technique with linear-shaped charge [8] offered the most promising features of being precise, controllable, and flexible to cut a through wall crack. Scoping tests were performed and confirmed that the cut by explosive was precise and quite uniform.

The schematic of flaw design is shown in Figure 9. With given pipe material and sizes, one calculates the critical flaw length, L_c, at testing temperature and pressure. The external flaw is then machined either below L_c or above L_c with a part-through wall condition. The remaining ligament thickness must be such that it does not lead to uncontrolled crack opening at testing temperature and pressure. Based on the remaining ligament thickness, another critical crack length ℓ_c can be calculated. With the line shape charge to generate a through wall crack greater than ℓ_c, unstable crack propagation will lead to through wall crack condition throughout the entire length of the initial flaw. Depending on whether that flaw is below L_c or above L_c, one can achieve the desired leakage or rupture conditions.

The test specimen length was determined such that the most critical transient phase of pipe rupture (depressurization corresponding to one cross-sectional area opening) is over before the reflected depressurization wave propagates back to the crack region. Assuming a two-phase fluid sound speed of ~400 feet/second and a critical transient time of ~15 milliseconds, the required pipe specimen length was calculated to be of order 12 feet.

3.3 Test Matrix

The text matrix is shown in Table 8. All pipes were of six-inch diameter and of either stainless steel or carbon steel. Material characterization tests were performed for each specimen. The range of machined flaws was such that it encompassed the important combinations of (a) subcritical flaw with a supercritical through wall crack within the flaw generated by linear-shaped charge, (b) supercritical flaw with supercritical through wall crack, and (c) supercritical flaw with subcritical through wall crack. Figure 10 illustrates the matrix of flaw configuration designs. The case of supercritical flaw with subcritical through wall crack was to study and verify that a small through wall crack embedded in a very long and deep part-through crack would not lead to any rupture but leakage only. The test matrix thus envelopes the most conservative conditions of leak and break of axial cracks.

Table 9 summarizes the flaw length calculations based on [7], using the actual measured material data. Pipe geometric dimensions are according to specifications.

3.4 Test Results

The test results of the six hot tests at LWR conditions together with that of a cold check-out test at pressure only (1300 psi ambient temperature) are discussed below.

(1) Crack Deformation Morphology

The check out test was of A106 carbon steel and had a supercritical flaw length of 17 inches. With a 4-inch linear shape charge to initiate a through wall crack at the center of the machined flaw, the crack propagated through the entire flaw and ran approximately 4 and 6.5 inches from each flaw end and then arrested. This resulted in an unsymmetrical crack opening area of about 64 in^2. The crack had abrupt circumferential turning right before arrest.

The first hot test performed was Test 1 given in Table 1. This was a 6-inch diameter, Schedule 80 carbon steel pipe with supercritical surface flaw (15.8 inches) and supercritical through wall crack initiation (4 inches). The unstable nature of the crack resulted in the total separation of the center section. Careful examination showed that the crack turned circumferential at the machined crack tip without any further propagation. This abrupt circumferential turning was the same as with the cold test, and the dynamic inertia of the crack opening associated with the stored energy release apparently led to the final crack separation configuration. A similar phenomenon was also observed for Test 5 of stainless steel pipe. The material and support location change, as well as flaw width (1/16 inch instead of 1/8 inch), did not change the unstable crack opening behavior. In this case, the crack propagated approximately 10 inches into the virgin pipe at each end before total separation.

Tests 2, 3, 4 and 6 are of stable crack configurations. Test 2, 3 and 4 had a subcritical machined flaw and a supercritical through wall crack initiated by shape charge. Test 6 had a supercritical flaw but a subcritical through wall crack as an initiator. As expected, no unstable propagation was observed. One notes that Test 6 is a repeat of Test 5 except that the shape charge was designed to give a subcritical through wall crack (1 inch). Since no crack propagation and opening occurred in this case, it shows that a sufficiently long through wall crack embedded in a long part-through flaw is needed before the pipe has any possibility of rupture. The unlikely nature of this to happen in actuality confirms that leakage will certainly be observed long before any potential of rupture.

(2) Reaction Force

Supports instrumented with load cells were placed underneath the test specimen to record jet thrust reaction forces due to pipe crack opening and system depresurization. These experimental measurements provided a direct assessment of the acceptance criterion given in [9]. The measured data are summarized in Table 10.

Preliminary analytical evaluation of thrust force was performed according to

$$T = KPA \tag{1}$$

where

P = system pressure prior to break
A = pipe break area
K = thrust coefficient.

According to ANSI 58.2 [5], for subcooled water without flashing, the acceptable value for K is 2 and for flashing jet, K is 1.35.

For the cold check out test, it was found that using the uniform system pressure prior to break was too conservative for the thrust evaluation. Even using local pressure in the pipe break region, the ANSI formula was still conservative by a factor about 2 since the K calculated based on experimental data is 1.07 instead of 2 as given in Table 10.

For Hot Test No. 1 with the use of system pressure, K was calculated to be 0.77. This K is much smaller than 1.35 and again shows the conservatism of ANSI value. Using the local pressure in the neighborhood of the crack region yielded K = 1.42, which is close to 1.35.

Tests 2 and 3 had small crack opening areas (5.6 in^2 and 2.4 in^2 respectively). The same simplified procedure was used to evaluate the jet thrust reaction force. The results are tabulated in Table 10. One observes that the discharge coefficient K based on the measured data was calculated to be ~3 while the value suggested in ANSI 58.2 for subcooled flashing fluid is ~1.4. This indicates that for the smaller opening case, the measured load is larger than the calculated and there is a factor of 2 difference. The exact cause of this discrepancy is not very clear. One postulate may be that the shape factor plays a very significant role in small opening conditions. More study is needed to resolve this question. The instrumentation functioning and other possible environmental factors also need to be checked out.

(3) Thermalhydraulic Conditions

As shown in Figure 8, extensive pressure and temperature gages were installed to monitor the thermalhydraulic state during the crack opening and the depressurization transient. Figure 11 shows the mass flux derived from the data and gives reasonable conditions.

(4) Crack Propagation Speed

The breakwires installed did not function properly. Premature debonding occurred due to high temperature and dynamic conditions. Some success was achieved in Tests 5 and 6. However, there was not any propagation in Test 6, and the overall quality of the data for Test 5 is very questionable.

4. Concluding Remarks

The ductile nature of pipe material makes pipe rupture unlikely. Plant operation experience provides sufficient evidence to substantiate this industry view. In fact, the stringent requirements on Class I pipe system design virtually eliminates the possibility of pipe breaks. German licensing has moved toward removing pipe whip and jet impingement considerations in the primary system. An NRC-sponsored study at Lawrence Livermore National Laboratory [10] may also recommend discarding guillotine-type pipe break assumptions in the design of PWR primary systems and remove the requirement of pipe whip restraint and other mitigative requirements. On behalf of the industry, the Atomic Industrial Forum is coordinating additional recommendations on more generic criteria that include other pipe sysem classes.

EPRI's pipe whip research provides important data and methods for more-realistic evaluation of pipe rupture dynamic effects in cases in which pipe rupture remains postulated. The limited pipe rupture and depressurization tests show that pipe will definitely be detected for leakage long before critical rupture condition can be reached.

5. References

[1] Atomic Industrial Forum (M. Edelman), letter to U.S. Nuclear Regulatory Commission (H. Denton), dated March 28, 1983.

[2] "Experimental Studies of Pipe Whip and Impact," Final Report submitted by CEA and FRAMATOME to EPRI, to be published.

[3] "Nonlinear Dynamic Analysis of Pipe Whip Tests," Draft Final Report submitted by Nutech Engineers, Inc. to EPRI.

[4] "ABAQUS-EPGEN: A General Purpose Finite Element Code," EPRI NP-2709, prepared by Hibbitt, Karlsson and Sorensen, Inc., October, 1982.

[5] "Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture," ANSI/ANS-58.2, 1980.

[6] "Pipe Rupture and Depressurization Experiments," Draft Final Report submitted by Wyle Laboratories to EPRI.

[7] Eiber, R.J., Maxey, W.A., Duffy, A.R., and Atterbury, T.J., "Investigation of the Initiation and Extent of Ductile Pipe Rupture," BMI-1866, Battelle Columbus Laboratories, July, 1969.

[8] "Specifications and Properties of Jetcord," Explosive Technology, Inc., Fairfield, California, January, 1980.

[9] "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," U.S. NRC, NUREG-0800, Standard Review Plan, Section 3.6.2.

[10] Tenth Water Reactor Safety Research Information Meeting, NUREG/CP-0041, Vol. 5, October, 1982.

Table 1

TYPE I TEST PARAMETERS

	L_1		α		L_2		R		δ	
Test	(ft)	(m)	(ft)	(m)	(in)	(m)	(in)	(m)	(in)	(m)
1	12	3.6576	0	0	8	0.2032	4.5	.1143	0	0
2	"	"	4	1.2192	"	"	"	"	0	0
3	"	"	0	0	"	"	"	"	2	0.0508
4	"	"	2	0.6096	"	"	"	"	2	"
5	"	"	0	0	"	"	"	"	8	0.2032
6	"	"	2	0.6096	"	"	"	"	4	0.1016
7	"	"	0	0	"	"	"	"	8	0.2032
8	"	"	2	0.6096	"	"	"	"	2	0.0508
9	"	"	0	0	14	0.3556	10.5	0.2667	8	0.2032
10	"	"	0	0	72	1.8288	4.5	.1143	2	0.0508
11	"	"	4	1.2192	"	"	"	"	"	"

Pipe sizes were 3 inch schedule 80 for all tests except for tests 7 and 8 (3 inch schedule 10).

Table 2

TYPE II TEST PARAMETERS

	L_1		L_2		δ	
Test	(ft)	(m)	(ft)	(m)	(in)	(m)
1	10.5	3.2	6	1.8288	60	5
2	"	"	"	"	"	"
3	"	"	"	"	"	"
4	"	"	"	"	"	
5	"	"	"	"	"	"

Pipe sizes were 3-inch schedule 80. The major parameter of Type II tests was concrete target given in Table 3.

Table 3

CONCRETE SLAB PARAMETERS

Test Number	Slab Thickness mm	Slab Thickness in	Concrete Compressive Strength 28 Day MPa	Concrete Compressive Strength 28 Day psi	Reinforcement Density kg/m^3	Reinforcement Density lb/ft^3	o Horizontal Rebars o Spacing mm (in)	Vertical Rebars
1	152.4	6	60.5	8775	150	9.36	HA 8 100 x 100 (3.9 x 3.9)	HA 6 each node
2	76.2	3	44.8	6498	150	9.36	HA 6 100 x 100 (3.9 x 3.9)	HA 6 each node
3	114.3	4.5	44.9	6512	150	9.36	HA 6 60 x 60 (2.4 x 2.4)	Ø 6 each node in the central square 1 m x 1 m (3.3 ft x 3.3 ft)
4	114.3	4.5	27.3	3960	150	9.36	HA 6 60 x 60 (2.4 x 2.4)	Ø 6 each node in the central square 1 m x 1 m (3.3 ft x 3.3 ft)
5	114.3	4.5	27.3	3960	100	6.24	HA 6 90 x 90 (3.5 x 3.5)	Ø 6 each node in the central square 1 m x 1 m (3.3 ft x 3.3 ft)

HA = French reference for high grip steel rebar
Ø = French reference for mild steel rebar

Table 4

PRICIPAL RESULTS FOR WHIP PHASE

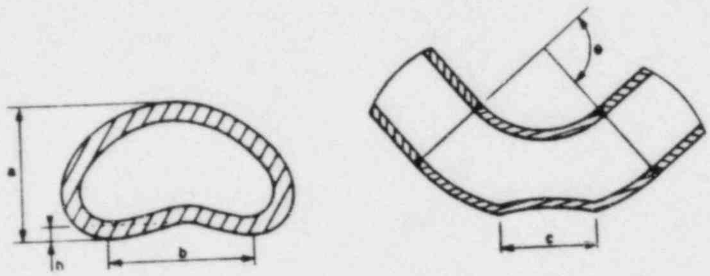
	Test Number	Gap (m)	Gap (inch)	Vertical Length (m)	Vertical Length (inch)	Pipe Whip Duration (ms)	Impact Velocity m/s	Impact Velocity ft/s
Type I Tests	1	0	0	0.2032	8	*	*	*
	2	0	0	"	"	*	*	*
	3	0.0508	2	"	"	3.7	**	**
	4	"	"	"	"	8.4	11	36.1
	5	0.2032	8	"	"	10.0	**	**
	6	0.1016	4	"	"	11.0	**	**
	7	0.2032	8	"	"	6.7	**	**
	8	0.0508	2	"	"	7.4	**	**
	9	0.2032	8	0.3556	14	14.2	22.5	73.8
	10	0.0508	2	1.8288	72	7.7	13	42.7
	11	"	"	1.8288	"	15.1	9	29.5
Type II Tests	1	1.524	60	1.8288	72	69.0	35	114.8
	2	"	"	"	"	65.0	**	**
	3	"	"	"	"	63.0	50	164.0
	4	"	"	"	"	76.0	33	108.3
	5	"	"	"	"	63.0	50	164.0

* For tests 1 and 2 there was no gap
** The recorded film is not exploitable because smokes ejected by the explosive cord came into the camera screen field.

Table 5

PRINCIPAL RESULTS FOR IMPACT ON RIGID TARGETS

Test	Maximum Impact Force		Impact Duration	Residual Crushing	
	10^4 N	Kip	ms	mm	in
1*	/	/	/	/	/
2**	1.8	4.1	6.	4.3	0.169
3	23.2	52.1	0.9	9.8	0.386
4	8.5	19.1	0.8	11.2	0.441
5	38.5	86.5	0.8	17.9	0.705
6	11.8	26.5	0.8	18.4	0.724
7	28.4	63.8	1.0	18.2	0.717
8	5.5	12.4	2.0	not available	
9	33.6	75.5	0.9	11.8	0.465
10	20.4	45.9	1.2	10.2	0.402
11	6.5	14.6	0.9	7.9	0.311



* For this test, the pipe was supported at the elbow. There was no pipe whip. The recorded force applied to the target was the jet reaction.

** For this test, there was no impact on the target (zero initial gap between the pipe and the target). The pipe was supported by the target in the straight pipe region. The given reduction in diameter is then only due to ovalization.

Table 6

REACTION FORCES ON SLAB SUPPORTS

Test	Maximum Force*		Time
	10^4 N	Kip	ms
1	45	101.2	15
2	28	63.0	28
3	45	101.2	1[illegible]
4	40	89.9	16
5	40	89.9	14

* Sum of all the load cells.

Table 7

GEOMETRY OF THE PIPE IMPACTING ZONE

	a		b		c		d		θ
Test	mm	in	mm	in	mm	in	mm	in	o
1	56.1	2.21	110	4.33	8.	0.31	211	8.31	107
2	65.5	2.58	61	2.40	4.6	0.18	124	4.88	85
3	54.8	2.16	82	3.23	10.2	0.40	169	6.65	103
4	61.2	2.41	71	2.80	8.2	0.32	150	5.91	95
5	57.4	2.26	77	3.03	9.4	0.37	168	6.61	97

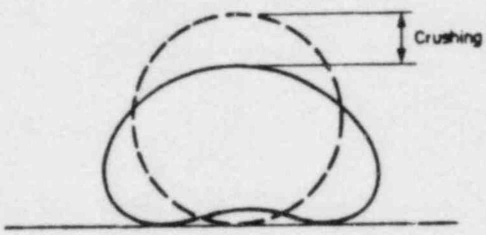


Table 8

TEST MATRIX†

Test Order	Pipe				Flaw*		Fluid		Crack Initiation**	
	80	160	C.S.	S.S.	S.	L.	BWR	PWR	S.	L.
check-out	x		x			x				
1	x		x			x	x			x
2	x		x		x		x			x
3	x			x	x		x			x
4		x		x	x			x		x
5	x			x		x	x			x
6	x			x		x			x	
7***		x		x		x		x		x

† All tests are performed use 6" pipes.

* S = Short < L_c; L = Long > L_c.

** Crack initiation is through wall crack initiated at the center of flaw using linear shape charge. S = Short < ℓ_c; L = Long > ℓ_c.

*** This one was planned but has not been performed.

Table 9

TEST CONDITIONS AND FLAW SIZES

Test Parameters					Material Properties		Flaw Sizes		Shape Charge
Test	Pipe Sch	Temp (°F)	Press (psi)	Material	σ_{flow} (ksi)	σ_{yield} (ksi)	Length (in)	Remaining Ligament (in)	Length (in)
check-out	80	70	1300	C.S.	57.0	52.7	17.0*	0.086	4†
1	80	530	1300	C.S.	61.0	49.4	19.5*	0.086	4†
2	80	530	1300	C.S.	61.0	49.4	11.7**	0.086	
3	80	530	1300	S.S.	38.0	25.8	6.8**	0.128	3†
4	160	540	2200	S.S.	34.1	21.1	8.4**	0.215	4†
5	80	530	1300	S.S.	38.0	25.8	11.4*	0.128	4†
6	80	530	1300	S.S.	38.0	25.8	11.4*	0.128	1††

*, † Supercritical

**, †† Subcritical

Table 10

REACTION FORCE

Test	A_{flaw} in2	tr Ms	(F1&F2) lbs	P_{tank} psi	P_{local} psi	T_{water} °F	K_{ANSI} (P_{tank}, T_{water})	$K = \frac{F}{P_{local} A_{flow}}$
check-out	64.	170	48,200	1300	700	70	2.	1.07
1	52.14	50	52.000	1300	700	530	1.35	1.42
2	5.6	60	20,500	1300	1300	530	1.35	2.8
3	2.4	60	9,400	1300	1300	530	1.35	3.01

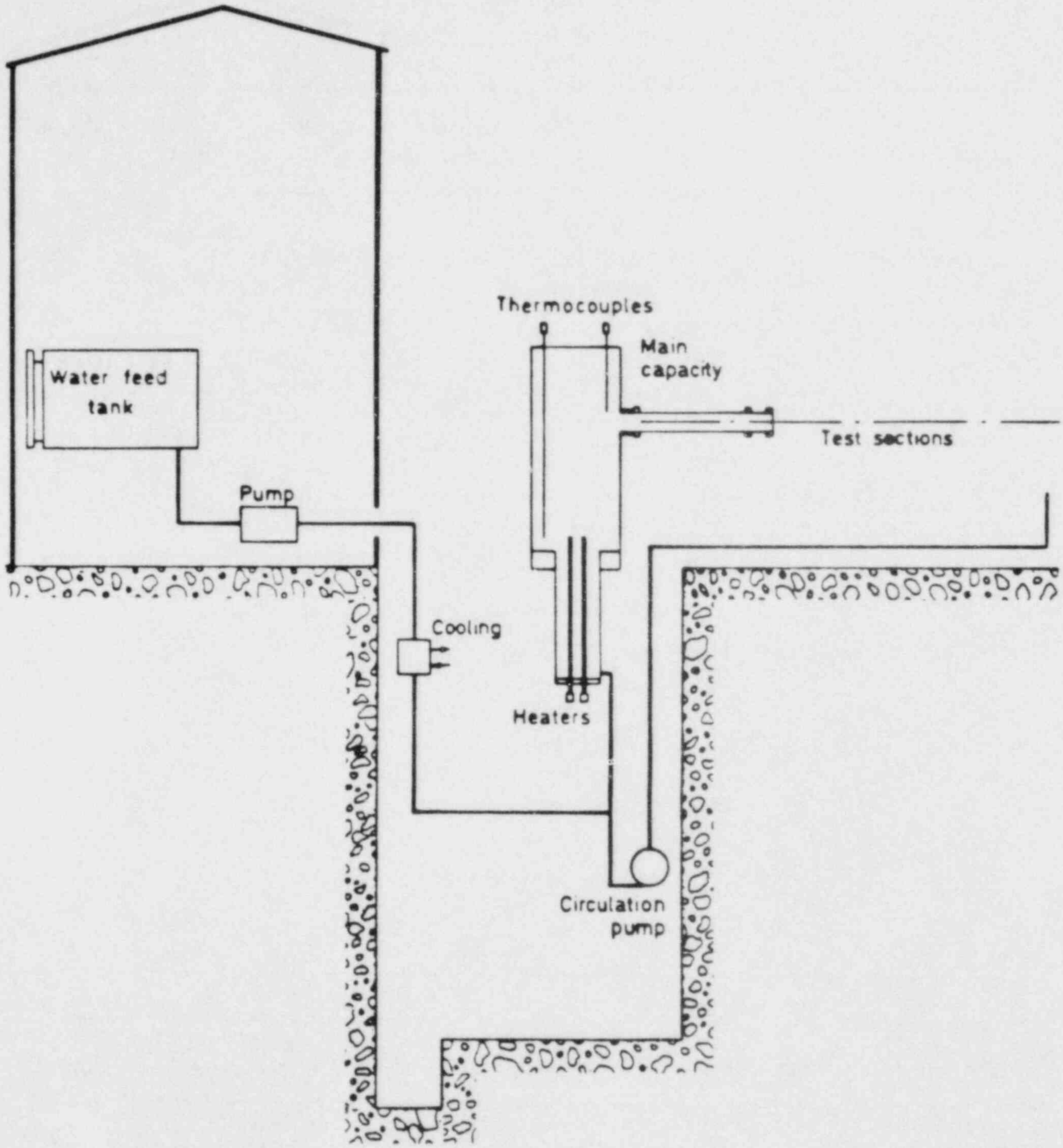


Figure 1. General layout of AQUITAINE II facility.

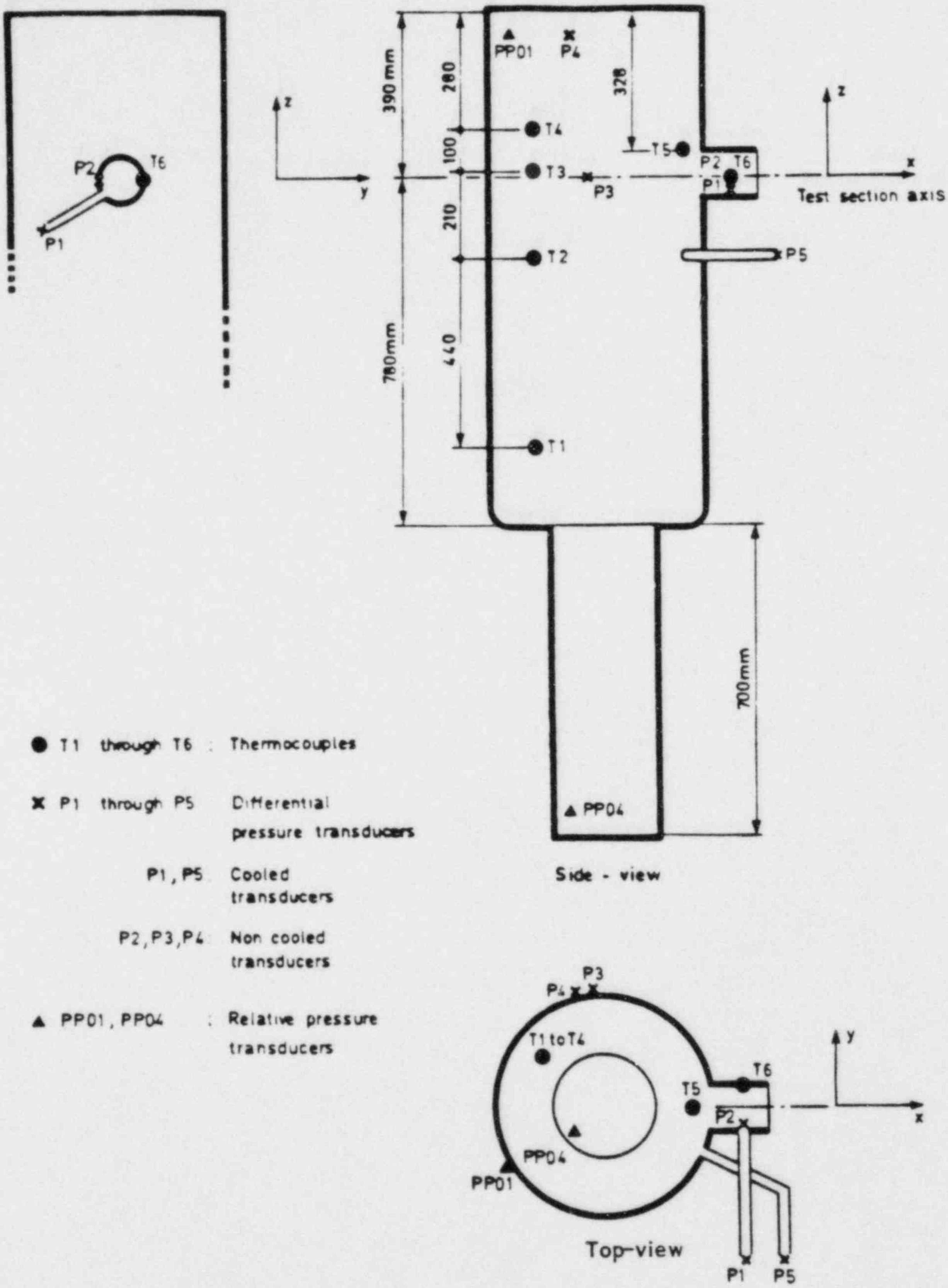
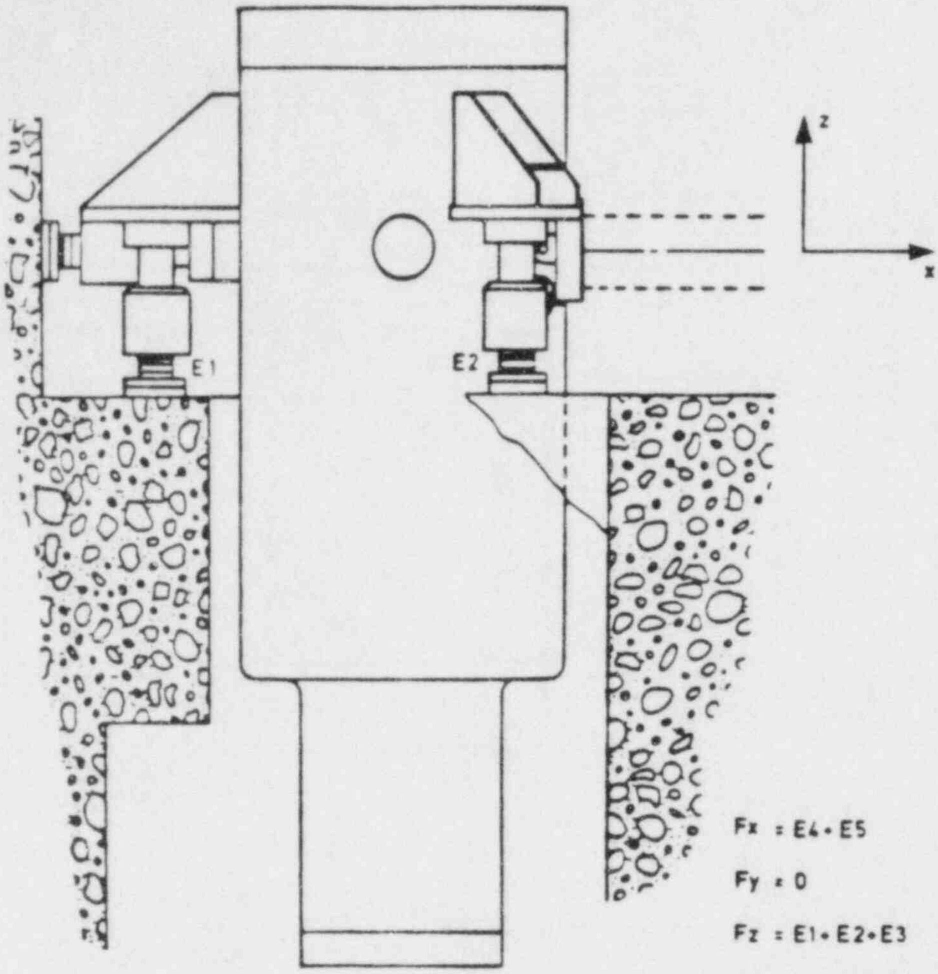


Figure 2. Position of transducers in AQUITAINE II vessel.



$$F_x = E4 + E5$$

$$F_y = 0$$

$$F_z = E1 + E2 + E3$$

$$M_x = -0.463\ E2 + 0.463\ E3$$

$$M_y = 0.535\ E1 - 0.267\ E2 - 0.267\ E3$$

$$M_z = -0.2\ E4 + 0.2\ E5$$

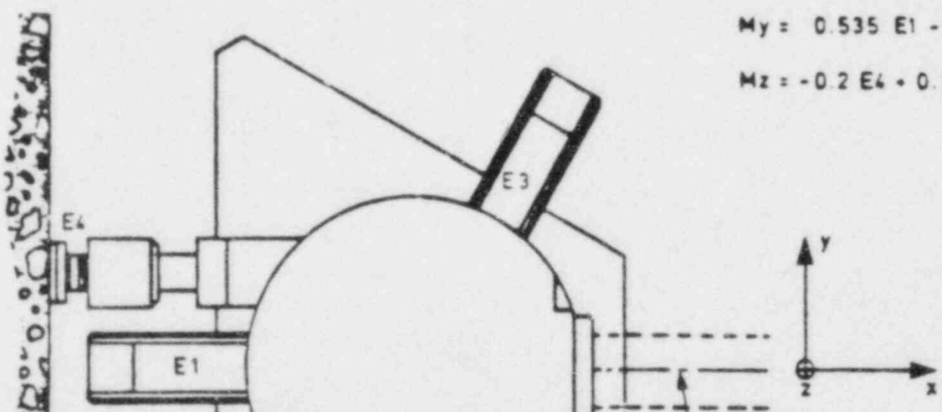


Figure 3. Position of support load cells.

T1 -- Temperature in the fluid
T2 -- Temperature on the wall
p1 -- Pressure in the fluid

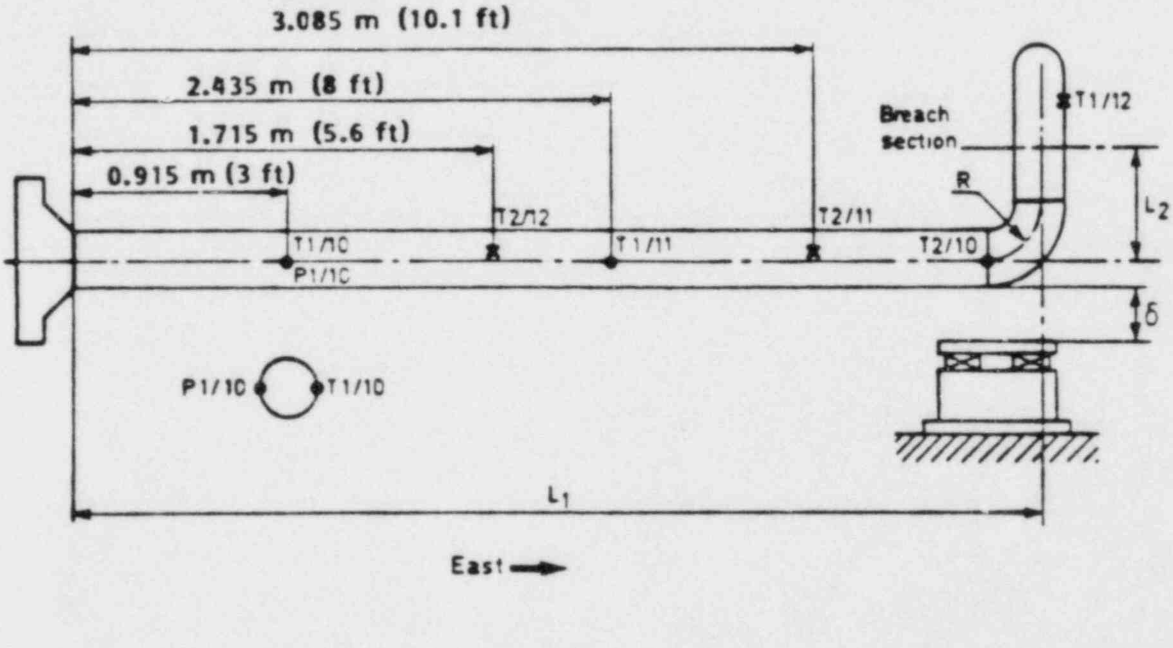


Figure 4. Typical test section.

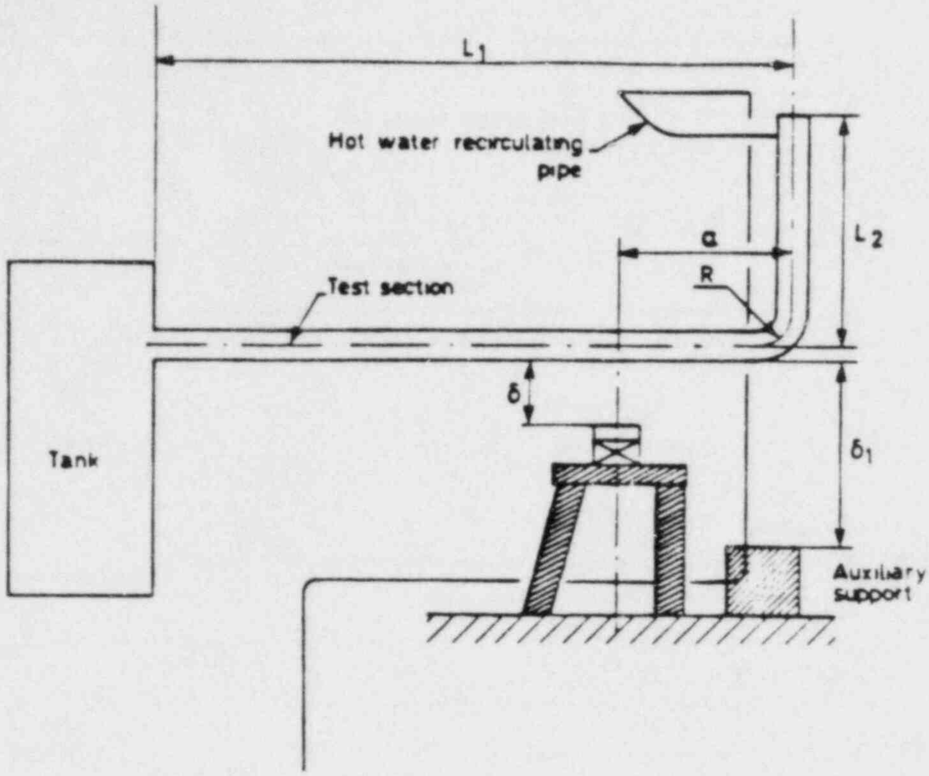


Figure 5. Typical sketch of type I test.

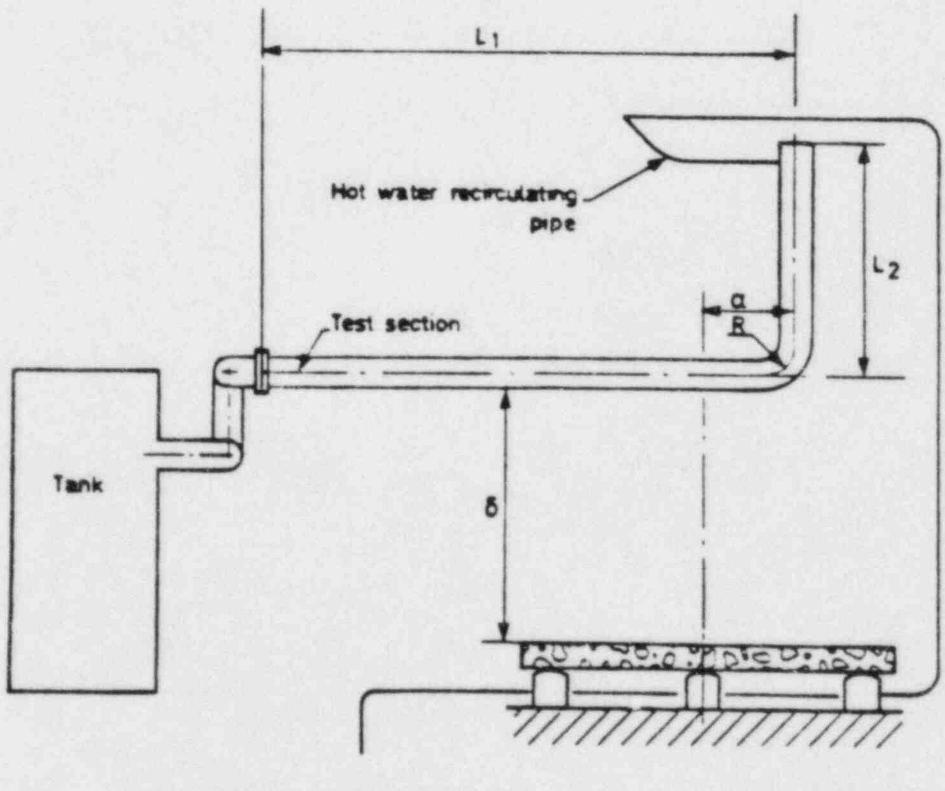


Figure 6. Typical sketch of type II test section.

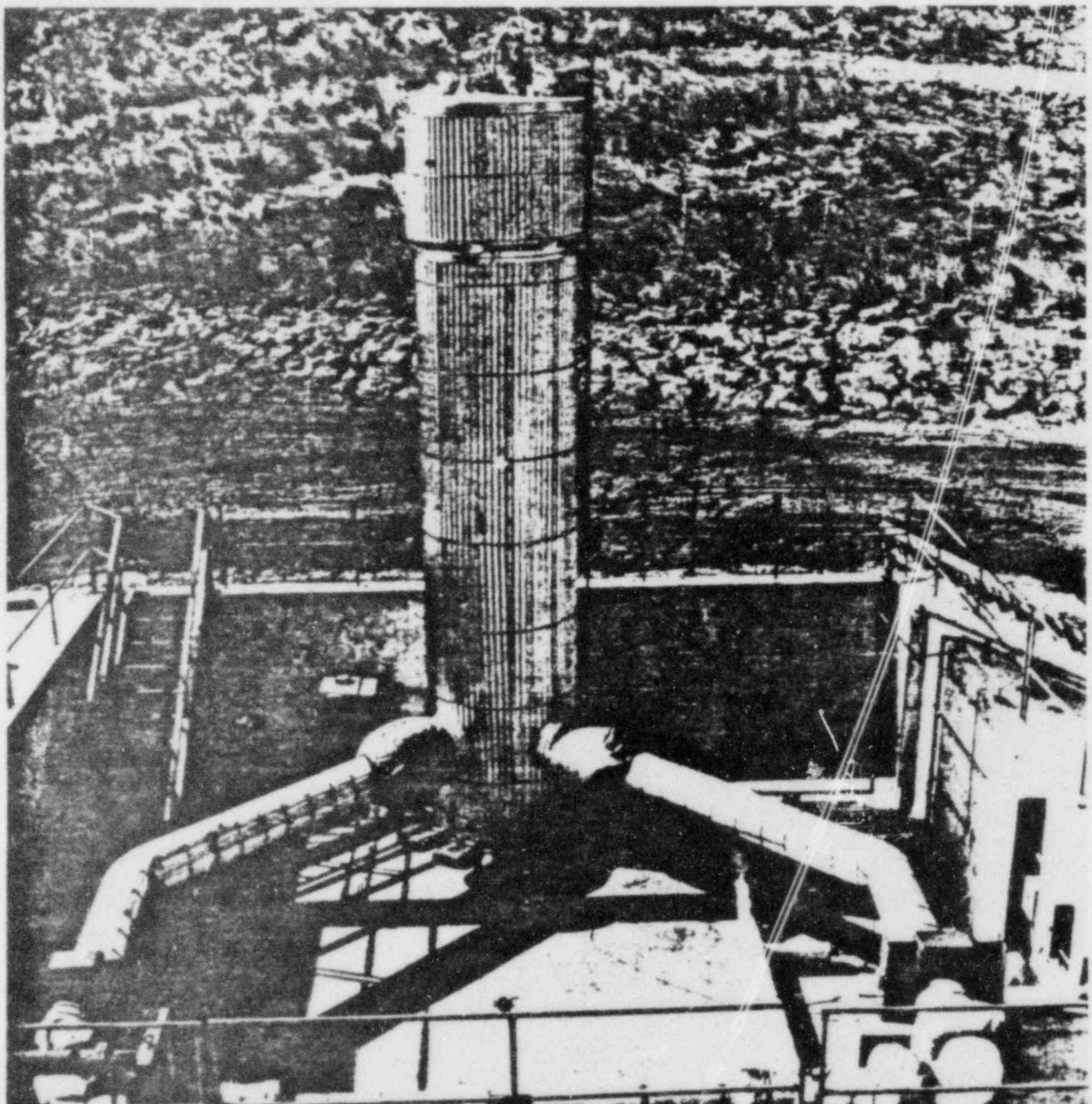
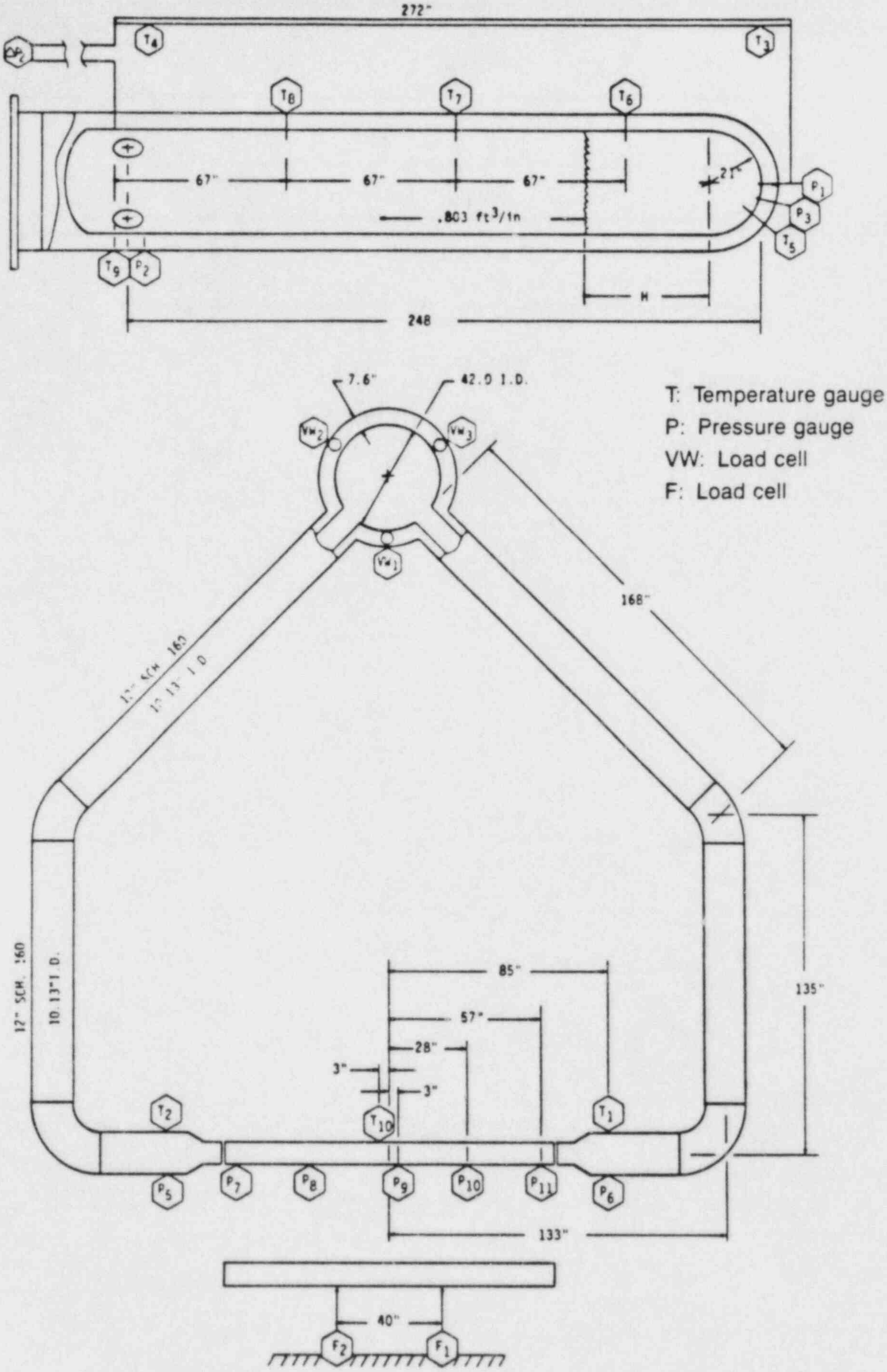


Figure 7. Test facility.



T: Temperature gauge
P: Pressure gauge
VW: Load cell
F: Load cell

Figure 8. Instrumentation layout.

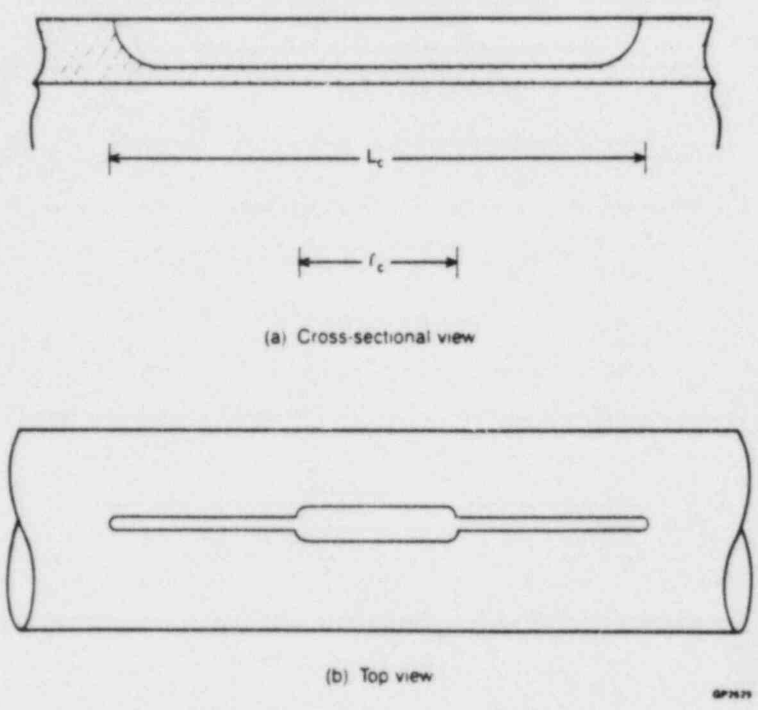


Figure 9. Schematic illustration of flaw design.

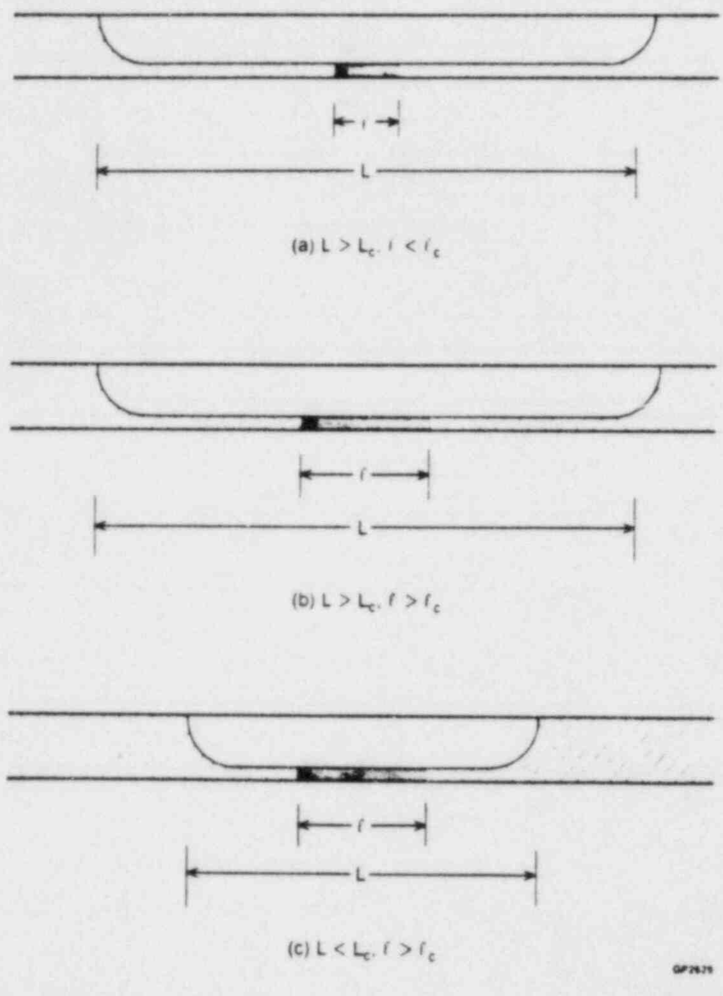


Figure 10. Matrix of flaw configuration designs tested.

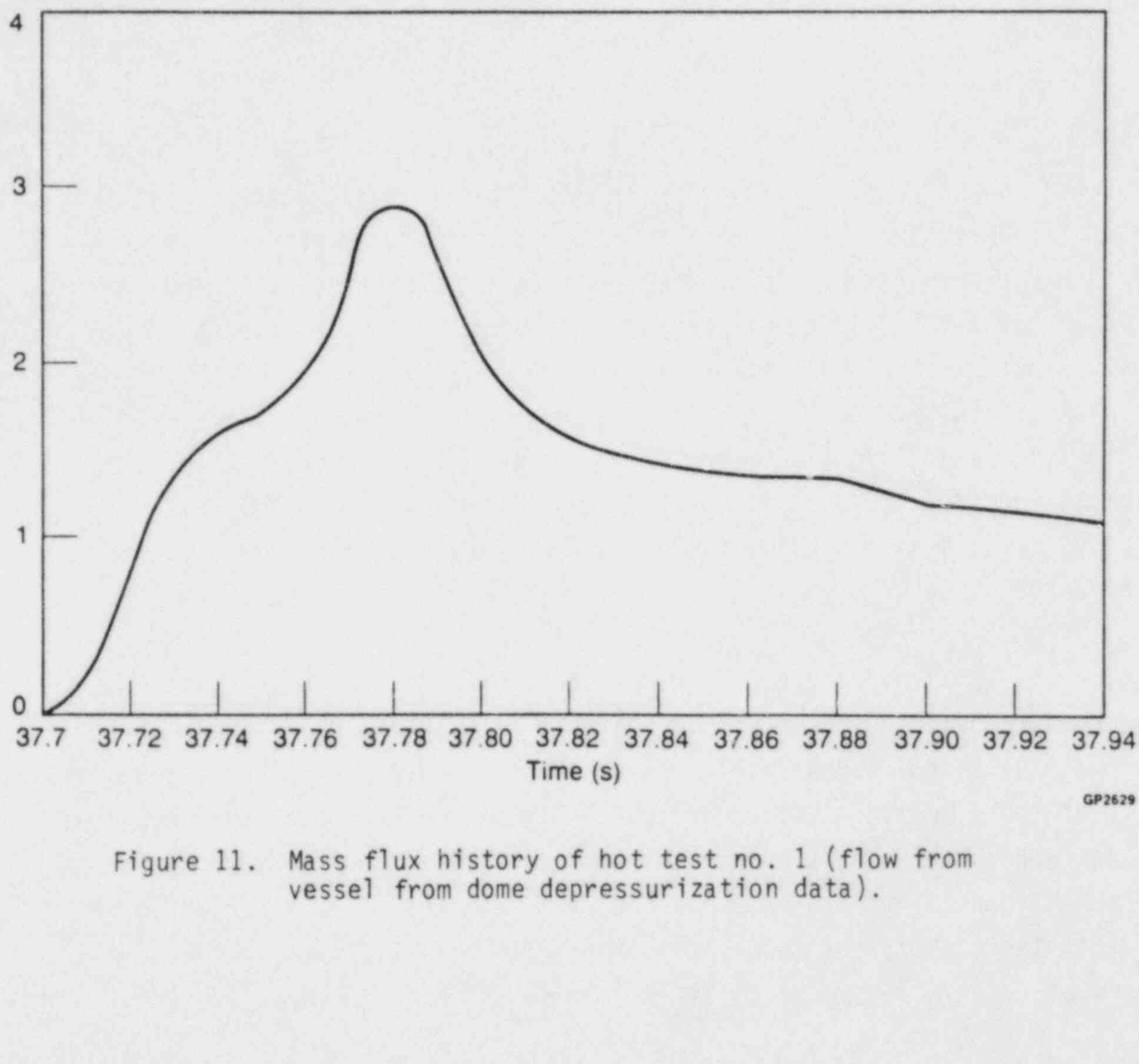


Figure 11. Mass flux history of hot test no. 1 (flow from vessel from dome depressurization data).

DESIGN TOOLS FOR COMPUTER-GENERATED DISPLAY OF INFORMATION TO OPERATORS

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The method of communicating process information to the operator in a nuclear power plant control room is presently in a state of transition; the hardwired instruments currently in use are slowly being supplemented and may eventually be replaced by information presented via computer-generated display systems. In the near term, this transition has been accelerated, since many utilities plan to use one or more visual display units to satisfy the regulatory requirement for a safety parameter display system (SPDS). In the longer term, nuclear industry vendors are marketing advanced control room designs which rely on multiple visual display units as the primary mode of communicating information to the operator.

Over the last several years, the Electric Power Research Institute has conducted research aimed at providing utility industry designers tools to meet the new design challenge created by the introduction of computer-generated displays. Two broad areas of research have been pursued: one has focused on using the computer to provide operators with more "intelligent" information about the plant and the other has focused on designing effective displays to communicate this information to the operators. This paper summarizes results achieved in these two areas.

TOOLS FOR DEVELOPING INTELLIGENT PROCESS INFORMATION FOR OPERATORS

The technology for providing intelligent displays in nuclear power plants has evolved from the computer-based process monitoring systems which were developed in the late 1960s and 1970s. Then, as now, digital data acquisition systems provide a direct plant process link to a host computer which is tied to colorgraphic displays used by operators. The major difference with these early process monitoring systems, however, is the complexity and sophistication of applications routines which operate on plant data to synthesize "new" information about the status of the plant. Rather than simply providing point status indication on a graphic display, intelligent process information systems support various levels of on-line analysis to simplify operator's status assessment and decision-making tasks. Modern process information systems are built on layers of supporting software or "tools" which can be used to interface multiple applications routines, or adapt the software to various operating systems environments. Considerable R&D effort has been spent by EPRI over the past several years to develop software tools for intelligent process information systems in nuclear power plants. The following describes some of these tools and general principles being used in their formulation.

1. Signal Validation. At the very root of any computerized information systems hierarchy is the quality of the input data received from the plant. In cooperation with a group of nuclear utilities, a computerized signal validation technology is being developed and demonstrated at a host utility that represents a major advance in validation capability. Based on the parity space technique (1), this validation methodology can employ direct and analytical signal redundancy to detect multiple sensor failures. A multidimensional decision space can be constructed from direct measurements, or pseudo-measurements which are obtained using a physical model of the process (see Figure 1). The decision estimator operates on a "parity space vector" in multidimensional space to: quality tag each signal; generate a weighted "best estimate" of the target variable. It is important to note that these techniques may be used in lieu of or in combination with simpler signal validation techniques such as limit checking or voting schemes. Judicious application of the technology should provide computer validated information which is far more reliable than hardwired control board indications.

The software design methodology being used in the signal validation project (RP2292) is shown in Figure 2. The Critical Safety Funtion (2) parameters which form the basis of many Safety Parameter Display Systems (SPDSs) are matched to pre-packaged signal validation modules, which are configured into a generalized signal validation system (SVS) architecture. The end-result is a tool-kit that utilities can mix and match to meet individual signal validation requirements and needs (see Figure 3). Although the SPDS is the immediate application objective in this project, the base technology should be integrated with the full range of normal and emergency plant information tools being developed by EPRI.

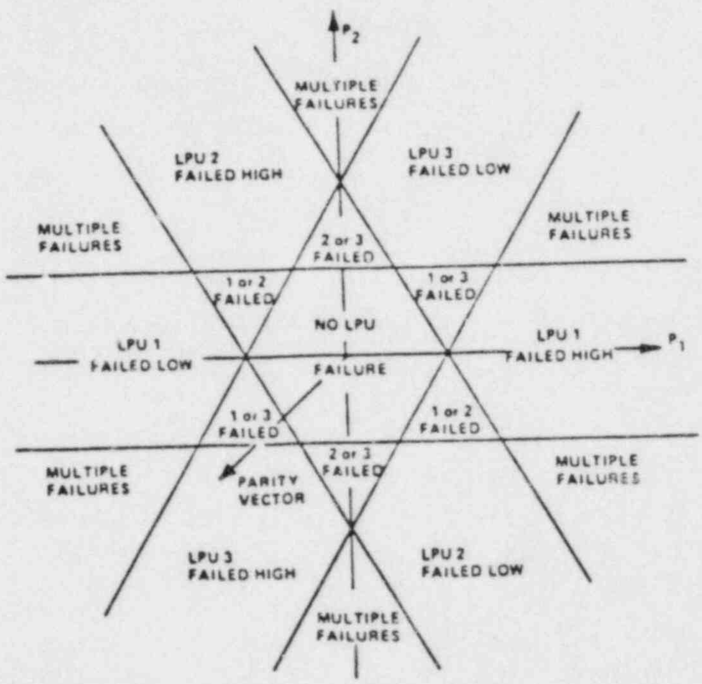

FIGURE 1. Parity Space with Decision Regions

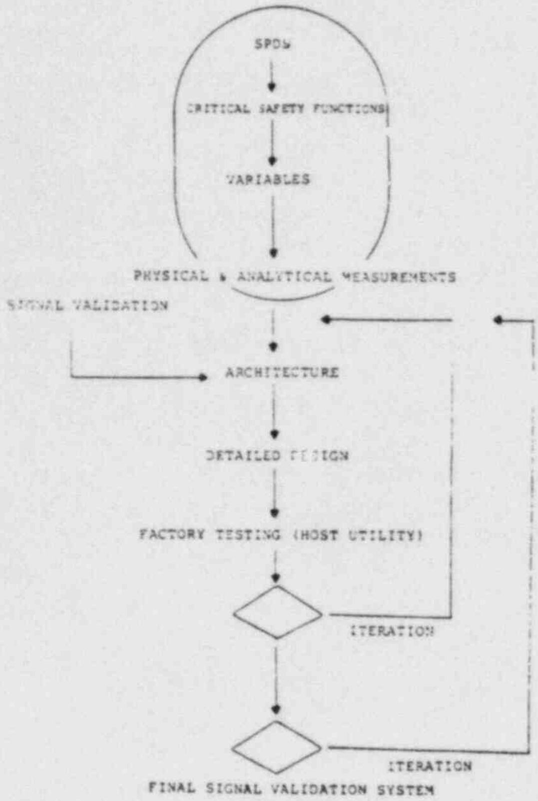

FIGURE 2. Signal Validation Software Design Methodology

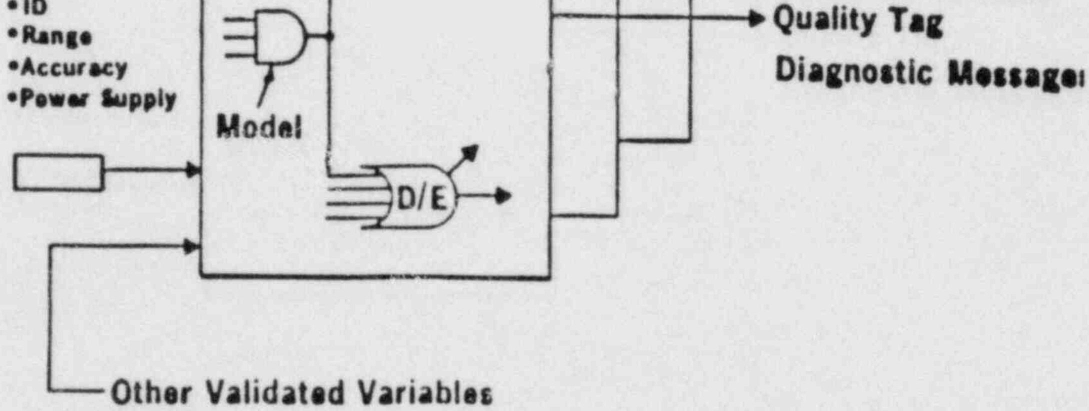

FIGURE 3. Signal Validation Tool Kit

2. Advanced Core Monitoring Framework. In parallel with the development of the SPDS has been a wholesale upgrade and replacement of plant process computer installations in nuclear power plants. Table 1 presents a typical comparison of original and replacement computer capabilities. Along with the substantial increase in computing power which is accompanying these retrofit activities is the increasing level of sophistication in core monitoring routines being applied. Core monitoring routines can provide on-line 2- and 3-dimensional predictions of power distribution, automatic checking of limiting conditions, on-line sensor calibration and diagnostics, exposure histories and statistical analyses. These developments have been beneficial, but the proliferation of software and fuel suppliers is leading to a hodge-podge of applications which do not fit within any given computer system.

TABLE 1. Computer System Hardware Comparison (Typical)

	ORIGINAL PLANT PROCESS COMPUTER	REPLACEMENT PLANT PROCESS COMPUTER
INPUT		
	SCAN RATE 100 PTS/SEC	SCAN RATE 6,000 PTS/SEC
	ANALOG POINTS 250	ANALOG POINTS 1000
	DIGITAL 600	DIGITAL POINTS 2000
	MONOLITHIC FRONT END	DISTRIBUTED, REMOTE MUX
CENTRAL PROCESSING UNIT		
	RANDOM ACCESS MEMORY 96 KBYTES	6 MBYTES
	WORD SIZE 24 BITS	32 BITS
	SPEED 5,000 FLOPS	0.3 MFLOPS
	MAX PROGRAM SIZE 24 KBYTES	600 MBYTES
BULK MEMORY		
	DRUM 1.1 MBYTES	DISK 600 MBYTES
OUTPUT		
	TREND RECORDERS (3)	6 FULL COLOGRAPHIC CRT's
	PAPER TAPE	MAGNETIC TAPE UNITS (2)
	LOW SPEED TYPERS (2) 17 CHAR/SEC	HIGH SPEED LINE PRINTERS 2,200 CHAR/SEC
	DIGITAL DISPLAYS (6)	4 STANDARD CRT's
CONFIGURATION		
	NON-REDUNDANT	DUAL REDUNDANT

The Advanced Core Monitoring Framework (ACMF) is being developed by EPRI in cooperation with a group of BWR utilities and fuel vendors (General Electric and EXXON). This project (RP1442) represents more than a collection of software tools in the sense that ACMF is a 'super' operating system which provides a "standardized" environment for core monitoring applications modules (3, 4). Refer to Figure 4. The basic framework features a system administrator which schedules jobs, handles external process computer requests, and generates operations logs. A display drivers library, standard display set and menu drivers supports control room operators and core performance analysis personnel. A graphic display of core power distribution is shown in Figure 5. Comprehensive data file management and system libraries are linked to the host computer operating system. In addition standard modules support statistical analyses, core exposure increments, plant data procurement, etc.

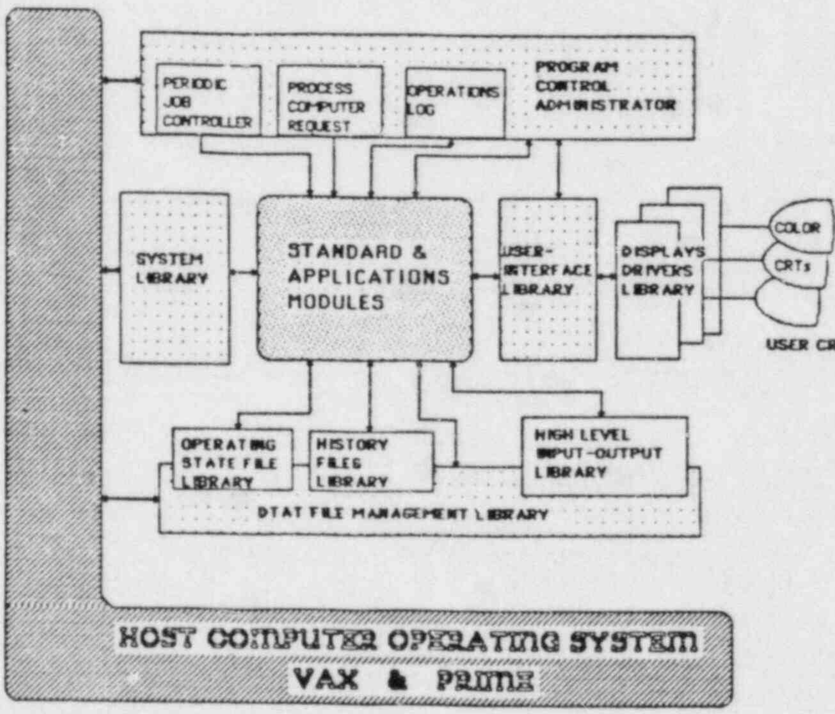

FIGURE 4. Advanced Core Monitoring Framework Software

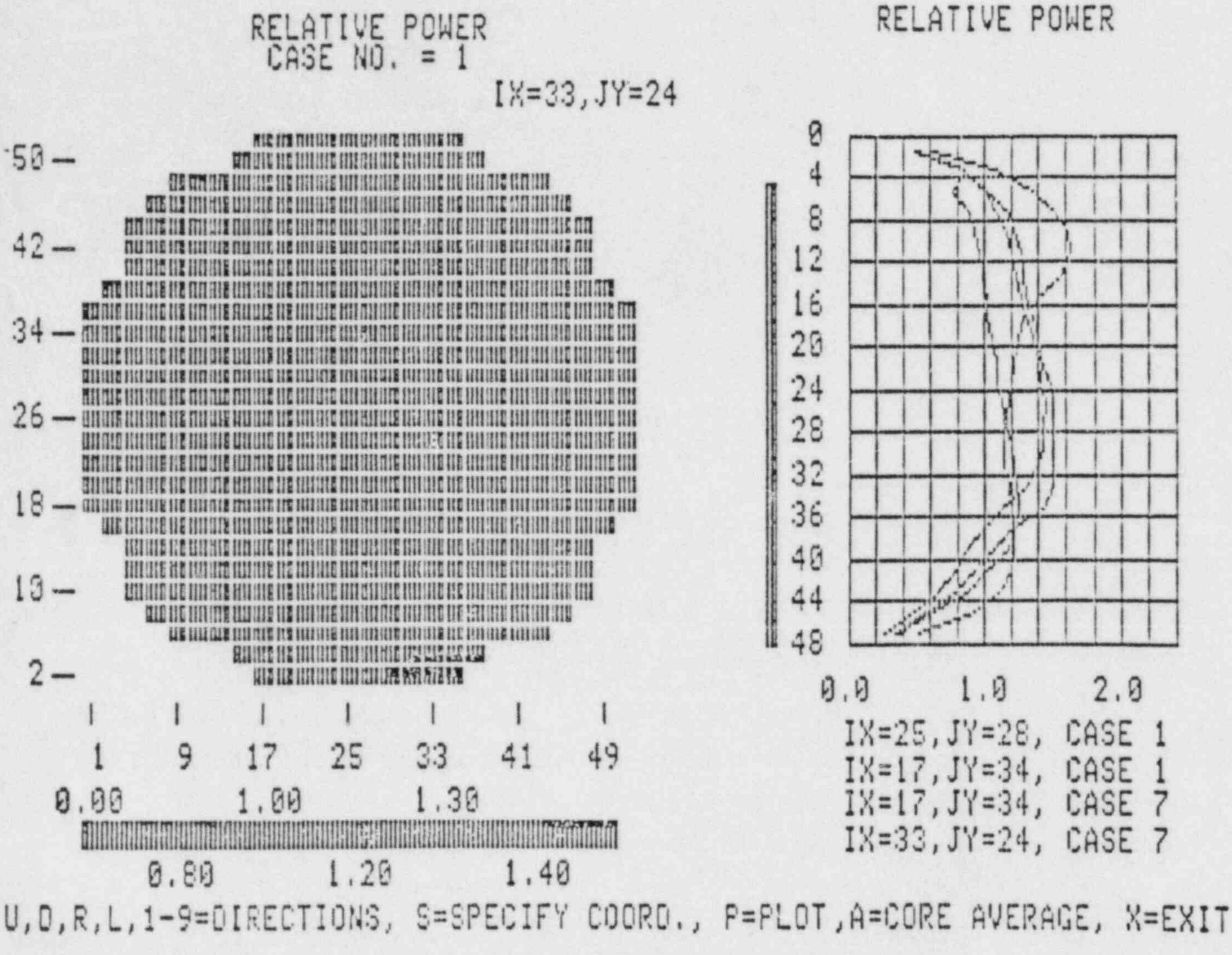

FIGURE 5. ACMF Standard Core Power Distribution Graphic Display

The ACMF software is "machine independent" in the sense that it can run on both DEC/VAX and PRIME mini-computers. This meets the portability requirement for software tools. But, the real advantage lies with its extensibility. Within ACMF utilities can replace, compare or add core monitoring modules from different suppliers without changing computer hardware or affecting user's interfaces. This is particularly important in situations where a utility is in the process of changing fuel suppliers. ACMF also permits a phased enhancement by permitting utilities to substitute core monitoring and accessory functions (e.g., TIPS, RWM and RPIS) as advanced applications become available from various software vendors.

3. Advanced Display Editor/Compiler. The highest level in the hierarchy of software tools is concerned with the user interface. Intelligent computer generated displays are critically dependent on the quality of software used to develop and drive them. The developer needs a convenient means to build and modify graphic displays and link to dynamic data bases without having to be concerned with the details of computer programming. This facility has been developed as part of the Disturbance Analysis and Surveillance System (DASS) project at EPRI during the past several years (5, 6). Recently, the advanced display editor/compiler has been upgraded extensively to support a variety of graphics terminals (Chromatics, Industrial Data Terminals, IBM-PC, etc.) with improved portability and maintainability features. The advanced display editor/compiler differs from other graphics development software because it is written in a high level language, is symbol-oriented, and provides direct links to dynamic data bases.

These software tools are pre-packaged routines that are much like spreadsheet or word processing programs which run on personal computers. The overriding advantage of employing a well-integrated set of software tools for process monitoring has to do with the significant developmental costs of validated computer software. Table 2 lists the costs of software development for various uses (7). For process monitoring software running in the neighborhood of 10,000 to 20,000 lines of the code the development costs may be anywhere from one to six million dollars. The practice of manually handcrafting applications-specific process monitoring software must give way to the development of software tools which can be used over and over.

TABLE 2. Effort and Cost to Product Software

	EPRI	Large Military Projects	Scientific Calculations (Ref 1-7)	Control of Expensive Processes (Ref 1-7)	Control Public Safety Processes (Ref 1-7)
Hours/Line of Code	2.1	3.5 - 7			
Cost/Line of Code	$60	$190	$10 - $40	$100 - $300	$300 - $700

The experience gained so far in the development of software tools for intelligent displays has pointed out the need for certain general design attributes. Specifically, software tools should be configurable, portable, extensible and maintainable. Configurable software can be applied to different uses without requiring the user to descend to the programming level to alter or make modifications. This is particularly important in the U.S. nuclear industry where considerable customization is needed to accommodate the various plant designs and operating practices. Portable software is needed to deal with the proliferation of computer types in nuclear plant process monitoring applications. Also, since the capital investment is generally much greater for computer software than hardware, there are enormous advantages in being able to transport software to replacement computer systems; there must be a way to decouple the software from fairly rapid computer hardware obsolescence. Extensible software is capable of enhancement by interfacing with additional applications routines or provide better functionality by substituting enhanced modules. This is important where a graded approach to installing intelligent process monitoring is desired. Maintainable software not only concerns configuration control and documentation, but the ease with which the software can be adapted to reflect changes in the plant process and computer operating systems designs.

The software tools represent the beginnings of a concerted EPRI effort to provide utilities with the capabilities for developing or adapting a variety of intelligent displays to serve both normal and emergency operating needs. Considerable work is needed to link the tools and provide a comprehensive tools package for general utility use. From a broader perspective, however, attention must be given to the understandability and usability of the displays for designated operations tasks. This latter concern is addressed by the human factors development and evaluation tools for graphic display systems which are described in the next section.

TOOLS FOR DESIGNING AND EVALUATING COMPUTER-GENERATED DISPLAYS

The ultimate goal of any control room information processing system, computer-based or otherwise, is effective communication with the human operator. To complement the heirarchy of tools described above, EPRI is also developing guidelines to assist utility designers and engineers in developing displays that communicate effectively with control room personnel. To date, a two-volume report has been published, "Computer-Generated Display System Guidelines," EPRI NP-3701, Volumes 1 and 2 (8, 9). Volume 1 provides a step-by-step procedure for designing displays and Volume 2 provides guidance on how to evaluate user interaction with displays and validate design trade-offs prior to actual plant implementation. Both volumes emphasize the human factors aspects of display design and evaluation and are developed for use by utility personnel to design and evaluate displays in-house or to more effectively monitor the progress of a vendor-supplied system from specifications to delivery and testing. Care was taken to structure the guides for use in backfitting computer-generated displays to control rooms of operating plants where hardwired, analog instrumentation is the primary means of information display and where any new display conventions must be compatible with those already in use.

Several key observations influenced the nature of these guidelines. First, early computer-generated display design efforts within the industry were to be applauded for their careful treatment of some human factors concerns, but few were comprehensive in their treatment of all important man-machine issues. This lack of comprehensiveness is not surprising since the relevant human factors design information, at the time of these early efforts, was spread across many diverse sources and had not been collated and filtered for the nuclear plant control application. More recently, the NRC has collated much of this data, and the guidelines reported here have taken advantage of that effort.

A second observation was that early design efforts were often inefficient in their use of utility design resources. Decisions regarding information displays were often revised and then revised again in the course of a design project. Each revision, of course, resulted in additional programming costs and, in some cases, project delays. Such false starts can be attributed, in part, to the lack of a clearly articulated design process that defines the sequence in which human factors concerns should be addressed to ensure both an effective and efficient design process. Existing human factors references focus more on definition of the issues and do not address the sequence in which these issues should be resolved in an ongoing design activity.

Another related observation was that early project decisions regarding the hardware or software approach to be taken overly constrained subsequent designer efforts to effectively communicate process related information to the operator. Decisions regarding information display were often tailored more to the performance requirements of the equipment selected that to the operator's real information needs as dictated by his assigned tasks. Again, there is a clear need to address design issues in the proper sequence and to emphasize early consideration of end user information requirements before other project decisions constrain effective man-machine communication.

It has also been observed that operational experience has not been effectively utilized in some early design efforts. Problems can be attributed to both under and over utilization of experienced operators. Omissions of key information, easily detectable by an experienced operator, have been discovered late in the design cycle of some projects which, unfortunately, had little or no early operational input. Yet in other projects, new, intelligent displays containing information unfamiliar to operators, but potentially beneficial, have been prematurely rejected because of negative operator opinion which had no basis in experience. Relevant operator expertise must be clearly defined and used appropriately.

The above observations had a significant impact on the tools EPRI sought to develop. The decision was made to formulate a display design process as opposed to generating new human factors design data. Clearly there are areas where design data are deficient and new data are required, but this need was deferred in lieu of an effort to put existing human factors data in a more useable form. Moreover, four aspects of the design process were to receive special attention: early definition of user information requirements, a comprehensive treatment of human factors issues relevant to the control room application, the proper sequence in which these issues should be addressed to ensure effective displays and efficient utilization of utility resources, and the appropriate use of operational experience.

One final observation also had a significant impact. The design process is imperfect in that trade-offs must be fashioned from a data-base that is strong in some areas yet weak in others. Therefore, design trade-offs should be validated prior to in-plant implementation to ensure that design objectives have been met. A second volume on display system evaluation was developed to aid utilities in conducting this validation.

Tools for Designing Displays

The increasingly significant role of digital computers and visual display units (VDU's) in power plant control rooms provides an opportunity for more intelligent display of information to control room operators. Apart from the obvious potential inherent in a larger data base and an expanded capability for logical processing of these data are the new communication tools available to the designer. He can, for example, select from a diverse set of picture elements -- bar charts, band charts, binary indicators, digital readouts, trend plots, and mimics -- to display a given information requirement, and he can combine different picture elements in the same display to convey multiple information requirements. Moreover, these picture elements can be further enhanced through the use of labels, color coding, symbol coding, highlighting and flash coding. Most important of all, the designer is no longer limited to the display of information at one spot on the control board but now has flexibility to combine information on individual displays as dictated by task requirements.

Successful exploitation of these new tools represents a considerable challenge to the display designer, perhaps even more so than with design of more conventional hardwired displays. A VDU is a serial display device. Unlike a hardwired instrument panel which provides parallel display of information, a VDU only shows at any one time a part of the information available in the computer's memory. Serial display, if carefully designed however, can be most effective since only the information required for the task at hand is displayed. Yet, it can be problematic if part of the required information is not displayed but retained in memory on another display, or worse yet, on several other displays. The designer, then, must make correct decisions about the information required for a task if the inherent advantages of a serial device are to be realized.

Likewise, the designer must make careful choices among available picture elements for effective communication of task-related information requirements. Typical hardwired displays in use permit the operator to extract different types of information -- deviation from setpoint, direction of change, rate of change, or a precise value -- from a single display of a given variable. More importantly, the operator can rapidly shift from extracting one type of information to another as the task dictates. Some picture elements available for use on computer-generated displays, while most effective in displaying one type of information (e.g., change of direction via a trend plot) are ineffective at providing other types of information (precise value from a trend plot). Designers of displays for VDU's then must carefully select picture elements that provide the type of information required for the task, and these elements must be grouped on individual displays so that the operator does not have to frequently call up different displays to obtain a different type of information.

The above issues are just two among the many that challenge the designer of computer-generated displays. If this challenge is to be met, the designer must carefully analyze the end users tasks and synthesize the results into meaningful displays by exploiting the capabilities of the computer while overcoming the inherent limitations of a serial display device.

Volume 1 of the EPRI guidelines (8) provides a process to aid the utility designer in meeting this challenge. Figure 6 shows an outline of the major steps involved in the process. The top-down approach begins with analysis of end user information requirements -- display system definition -- proceeds to systhesis of information requirements into pictures and parallel definition of constraints, the translation of pictures into programmable displays based on the contraints identified, and terminates with specification of hardware and software.

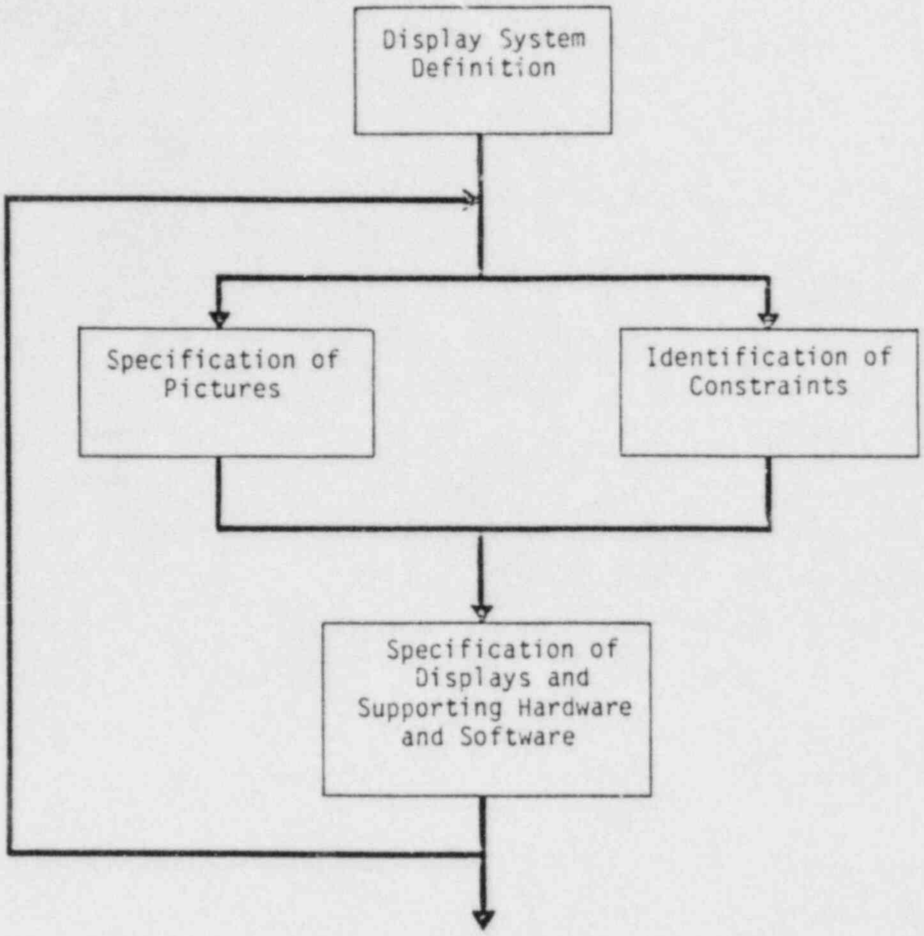

FIGURE 6. The Display Design Process

The initial step in the process follows a typical approach to systems analysis, albeit from the perspective of a VDU user. Information requirements are identified by systematically decomposing display system objectives into major functions, specific user tasks and finally the information required to perform these tasks. Volume 1 contains guidance on performing each of these steps with particular emphasis on the characteristics of information requirements needed for later selection of picture elements.

The analysis step is straight-forward for those applications in which a VDU is used to display information for existing operator or control room tasks. The appropriate written procedures, if carefully analyzed and supplemented with inputs from operators, provide the essential information needed for specification of information requirements. In those cases, though, in which implementation of the new system redefines operator tasks or creates new tasks, which is often the case with the implementation of computer-based operator aids, a much greater burden is placed on the display designer to translate his or someone else's design concepts into specific functions, tasks and information requirements. Clear documentation of this process is essential for such applications since it will be the basis of subsequent procedures and training and will also play a key role in system evaluation.

The next major step (see Figure 6) involves translation of information requirements into pictures. Pictures, as distinct from programmable displays, contain information needed to perform a function or task and as such are yet unconstrained by size, aspect ratio, resolution or other hardware implementation considerations. Pictures are synthesized via the steps listed in Table 3.

Table 3

SPECIFICATION OF PICTURES

Picture Synthesis Step	Volume 1 Guidance
Selecting Picture Elements	Procedure for matching task-related characteristics of information requirements to types of picture elements
Constructing Pictures	Guidance on assigning picture elements to a picture based on results of task analysis
Improving the Intelligibility of Pictures	Guidance in use of labeling, coding, and highlighting to improve communication
Evaluating Pictures	Procedure for using operations personnel to evaluate the understandability of individual displays
Specifying a Picture Structure	Guidance on specifying the relationship among pictures based on task analysis results
Specifying a Dialogue Structure	Guidance on selecting a structure (e.g., menu, command language, etc.) and device (cursor, keys, etc.) for user interaction with individual displays

The two major steps discussed thus far emphasize effective communication of process information to the display system user. The pictures which result from completing these steps should reflect user task requirements both in terms of the content of individual pictures and the relationship between pictures. The remaining two steps involve translation of these pictures into programmable, implementatable displays. The first of these -- Identification of Constraints (see Figure 6) -- involves specification of those aspects of system implementation that will draw boundaries around the acceptable design solutions. The particular issues discussed in Volume 1 are listed in Table 4.

Table 4

CONSTRAINTS IN RETROFIT DESIGN

1. Location of equipment
2. Electrical power for equipment
3. Operating environment for equipment
4. Availability of signals
5. Use of existing data systems
6. Use of existing display equipment
7. Use of a specific software language
8. Machine independence of software
9. Existing software
10. Future upgrades to the system
11. Impact on the operator's tasks
12. Maintenance requirements
13. Physical/data security
14. Application-dependent standards, quidelines, and regulations
15. Human factors standards, guidelines, and regulations

Once the issues in Table 4 have been resolved, the translation of pictures into scale drawings for later implementation as displays can be initiated. The steps involved in this process and the guidelines contained in Volume 1 to aid this process are summarized in Table 5.

Table 5

SPECIFICATION OF DISPLAYS

Display Specification Step	Vol[illegible]e 1 Guidance
Specifying display location and environment	Guidance on hardware configuration to optimize viewing distance, screen orientation, work station dimensions, ambient lighting, display/surround contrast and to minimize glare
Specifying, display hardware and software capabilities	Guidance on the selection of symbol size, interline spacing, figure/ground contrast, image polarity and refresh rate
Specify system performance characteristics	Guidance regarding user feedback, response time, data up date time and communication of system display malfunctions

While the process of synthesizing pictures and translating these into displays is driven by task-related information requirements, it is also shaped by knowledge of operator capabilities and limitations. Perceptual and cognitive abilities heavily influenced the choices associated with specification of pictures (Table 3) while operator visual abilities are emphasized in the translation of pictures into displays (Table 5).

The final step in the design process is to develop input for a hardware and software specification based on the scaled drawings of displays. Volume 1 contains detail guidelines on how to translate these prior design decisions into input for specifications. Results of a vendor survey are also included to aid utilities in making final hardware and software choices. Throughout the report, the type of expertise, including personnel with operating experience, needed to complete each step is clearly indicated.

Tools for Evaluating Displays

Display design has been correctly described as a series of tradeoff decisions in which requirements for communicating with the operator must be tailored to the realities of display implementation. Even though the display design process described in Volume 1 was developed to minimize the negative impact of such tradeoff decisions, the translation of pictures into useable displays is constrained by the availability of plant signals, the possible locations for a VDU in the control room, the hardware and software selected, and other constraints. Since some of these tradeoffs rely heavily on engineering judgment, especially given the weak character of human factors data in a few key areas, the adequacy of the ensuing design may be uncertain in some respects. Thus, to ensure that design objectives have been met, it is desirable to validate these tradeoffs prior to implementation of the system in the plant. To aid utilities in conducting these validations, Volume 2 (9) provides guidance in evaluating user interaction with display systems. It is structured for appliction following completion of the design process outlined in Volume 1 and once a working prototype of the system is available.

As with Volume 1, past experience also was a valuable guide in preparation of Volume 2. In particular, an approach was sought which would correct two problems inherent in earlier EPRI and related industry evaluation efforts: over reliance on expert opinion and the failure to address design issues in the most effective and efficient evaluation sequence.

Some earlier evaluations can be appropriately characterized as demonstrations in which experts, usually operators, viewed and reacted to displays. While such activities provide valuable insights, they do not provide definitive evidence that system objectives have been met. A more objective approach would be to confirm that a sample of users can perform the intended tasks while using the system. Such evaluations are, of course, more expensive which highlights the second concern -- both effective and efficient evaluations.

Some earlier efforts, which are to be praised for their attempt at objective evaluation, encountered difficulties because they attempted to assess overall system effectiveness without first determining that displays are optimally readable and easily understood. Therefore, these effectiveness evaluations had to be interrupted, at considerable costs, to correct problems related to display readability and understandability. These latter issues should be addressed first to ensure the most effective utilization of evaluation resources.

The approach to evaluation outlined in Figure 7 was formulated to correct these problems. Whereas the design process (Volume 1) is top-down, evaluation is bottom-up beginning first with an assessment of system compatability. The system is compatible to the extent that the users visual and anthropometrical capabilities have been accommodated by the choice of VDU location and environment and the detail features of the displays. The objective of compatability evaluation is to determine that users can easily detect, recognize and read display detail, discriminate the colors used, and reach the controls necessary to operate the display system. Since criteria regarding human visual capabilities and anthropometric limitations are well established, compatibility evaluation can usually be accomplished using a checklist which incorporates these criteria.[1]

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[1] If Volume 1 has been utilized in design, compatability evaluation is not required. It is recommended, however, in those cases where Volume 1 or an equivalent human factors guide has not been applied.

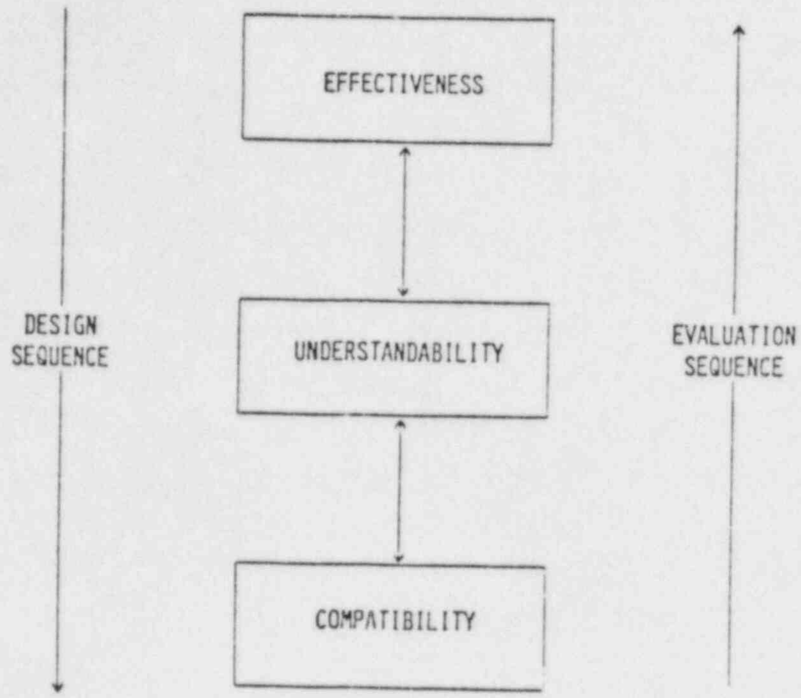

FIGURE 7. Levels of Design and Evaluation

Having established that displays are readable, the next concern is to ensure that the user can easily understand and interpret what is read. In contrast to assessment of system compatibility via checklists, understandability evaluation requires active user involvement with dynamic displays. One approach to such evaluation is to drive the displays with data recorded from the plant in a training simulator. After viewing the displays presented via this "part-task" simulator, the user can be queried to assess the degree to which the messages presented are comprehended.

The final evaluation issue is effectiveness. After, and only after, it has been established that the displays are readable and understandable, evaluation shifts to whether the display system supports the operator in performing those tasks circumscribed by the display system design objectives.

Thus, the tools necessary for effectiveness evaluation should not only provide capability for viewing of dynamic displays but should also provide capability for the operator to perform the required tasks while viewing these displays. This can be accomplished by either driving the displays with a model of the plant which accepts and is responsive to operator input in real time, or by installation of the display system at a full-scope training simulator. Clearly, the later approach provides a higher degree of fidelity in terms of simulating the operator's use of displayed information to formulate and implement control tasks. But in those cases where a simulator is unavailable, the model driven part-task simulation can provide valuable information. In either case, effectiveness is assessed in terms of whether user performance of simulated tasks meets some predetermined standard or represents an improvement over a pre-existing display situation.

Volume 2 contains detailed discussions on the three levels of evaluation issues -- compatibility, understandability, effectiveness -- and the tools -- paper methods, part-task simulation, full-scope simulation -- appropriate for each. There is also discussion on experimental methods appropriate for use in resolving evaluation problems at each level. Throughout, the importance of objective evaluation methods based on observations of users is emphasized as well as the need to address evaluation issues in the proper sequence.

The guidance contained in Volumes 1 and 2 was developed by a team of specialists including a nuclear engineer, a computer specialist and a human factors specialist. The skills and knowledge of the team members were supplemented by a literature review and a vendor survey. Most important of all, key inputs to the development effort were obtained by trial application of techniques and methods in the guidelines to evaluation of Disturbance Analysis and Surveillance System, a computer-based operator aid, developed under a separate EPRI project.

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BWR PIPE CRACK CONTROL USING HYDROGEN WATER CHEMISTRY: STATUS REPORT ON DRESDEN-2 PROGRAM

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ABSTRACT

One of the proposed remedies for intergranular stress corrosion cracking of stainless steel piping in BWRs is an alternative water chemistry called hydrogen water chemistry (H_2WC) that involves suppression of reactor water dissolved oxygen to ≤20 ppb via hydrogen injection to the feedwater in conjunction with control of conductivity to ≤0.3 μmho/cm. This paper describes progress to date on the in-plant verification of H_2WC at the Dresden-2 Reactor.

INTRODUCTION

A major environment-related materials performance problem encountered in the reactor coolant system of BWRs has been intergranular stress corrosion cracking (IGSCC) of sensitized austenitic stainless steel. IGSCC of sensitized material adjacent to welds in Type-304 and Type-316 stainless steel piping systems has been responsible for more than 400 cases of pipe cracking over the last ten years (1). Although these cracks are not thought to pose a major safety concern, inspections and repairs associated with pipe cracking have proved costly to the utilities and substantial R&D programs have been undertaken to understand the IGSCC phenomenon and develop remedial measures (1-3). Much of the early remedy-development work focused on alternative materials or local stress reduction, but recently, the effects of water chemistry parameters on the IGSCC process have received increasing attention in work funded by EPRI, the BWR Owner's Group, and the USNRC. A complete understanding of the interrelationship between BWR water chemistry variables and IGSCC of sensitized stainless steels has not yet emerged but some important features have been identified.

BASIS FOR AN ALTERNATIVE BWR WATER CHEMISTRY

Water itself is relatively innocuous towards stainless steels even at the relatively high temperatures involved in its use as a working fluid in electric generation plants. However, impurities that enter the water can make it an aggressive environment towards sensitized stainless steel. Oxygen is generated as a result of radiation in the core of a BWR during normal operation and this leads to a steady level of ~200 ppb dissolved oxygen in BWR water. Ionic impurities (salts) enter the water from sources such as makeup water and condenser leaks. These are controlled by condensate and reactor water cleanup systems which, however, can be costly to operate. The objective of the overall EPRI program on impurities in BWR water is to determine the effects of impurities on IGSCC so that guidelines for water quality to minimize pipe cracking can be formulated on a quantitative basis.

Impurities that exacerbate pipe cracking fall into two classes depending on their mode of action. The oxidizers, represented by oxygen dissolved in the water, increase the chemical driving force for the corrosion reactions; and, because the reactions are electrochemical in nature, ionic impurities, such as chloride ions, increase their rates by increasing the electrical conductivity of the water. Figure 1 represents current thinking about the synergism between the effects of the two classes of impurities on IGSCC (4). It is apparent that the contents of both classes of impurities must simultaneously be minimized to minimize the likelihood of IGSCC. This concept underlies an alternative BWR water chemistry, known as hydrogen water chemistry (H_2WC), which entails oxygen suppression via hydrogen addition to the feedwater in combination with conductivity control via optimized plant operational procedures. The remainder of this paper presents a status report on the development and in-plant verification of H_2WC.

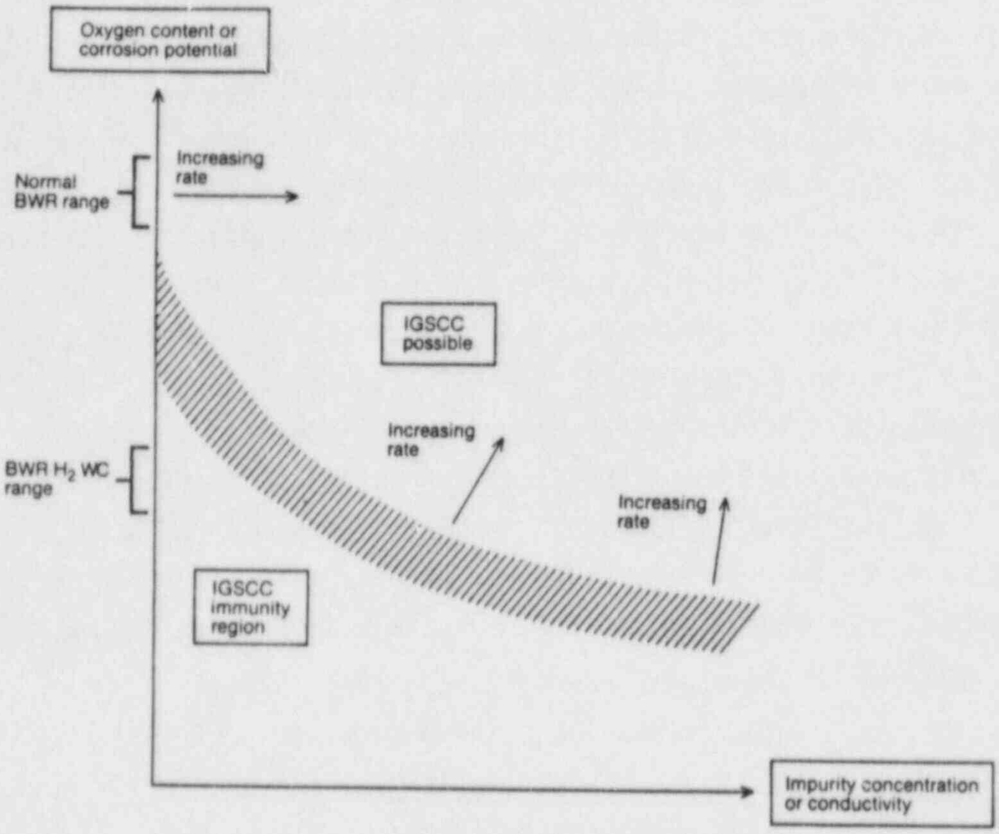

Figure 1. Schematic summary of the results of laboratory studies of the effect of impurities on IGSCC of stainless steels.

DRESDEN-2 HYDROGEN WATER CHEMISTRY VERIFICATION PROGRAM

Background

In 1977, the U.S. Department of Energy sponsored a program designed to identify additives suitable for oxygen suppression in the BWR, to determine their possible impact on various plant systems, and to select the best additive and test bed. The additive selected was hydrogen, and the demonstration plant selected was Commonwealth Edison Company's Dresden-2 reactor in Morris, Illinois.

In 1982, General Electric, DOE, Commonwealth Edison and EPRI participated in a one-month hydrogen addition feasibility demonstration in Dresden-2 (5). This one-month demonstration included in-reactor stress corrosion cracking (SCC) tests on furnace-sensitized stainless steel and low alloy steel specimens and electrochemical potential measurements on various BWR structural materials. The results showed that the coolant oxygen levels in the GE designed BWR can be suppressed to below 20 ppb during power operation by adding practical amounts of hydrogen to the feedwater, on the order of 1.5 ppm. Reactor water having an oxygen content of 20 ppb and a conductivity of 0.30 μmho/cm was shown to be insufficiently aggressive to promote either IGSCC in sensitized stainless steel or transgranular stress corrosion cracking of pressure vessel steel in short-term, slow strain-rate tests. At low oxygen levels, the quantity of N-16 in the steam increased as expected, but the consequent four to five-fold increase in steam line gamma radiation was not found to be a major problem in Dresden-2. No other significant adverse effects of H_2WC were identified in the relatively short-term demonstration.

The next step in the H_2WC program, a long-term verification over two or three 18-month fuel cycles started at Dresden-2 in April of 1983 under EPRI funding. It involves extensive monitoring of fuel and core materials behavior and continued evaluation of plant structural materials behavior in longer term tests. The water chemistry goals are ≤20 ppb oxygen and a reactor water conductivity of ≤0.3 μmho/cm.

The following paragraphs summarize the results to date on this program, problems encountered and their resolution, and system upgrade activities currently under way. This is an update of an earlier paper on this program presented at the 1984 American Power Conference (6).

Plant Operations

The hydrogen addition flowsheet is shown in Figure 2. High purity (99.999%) hydrogen is purchased as gas which is delivered in tank trucks carrying 115,000 or 50,000 scf at 2400 psi. These tanks supply hydrogen at about 200 psi through a 1/2" carbon steel line into a flow control panel in the condensate pump room.

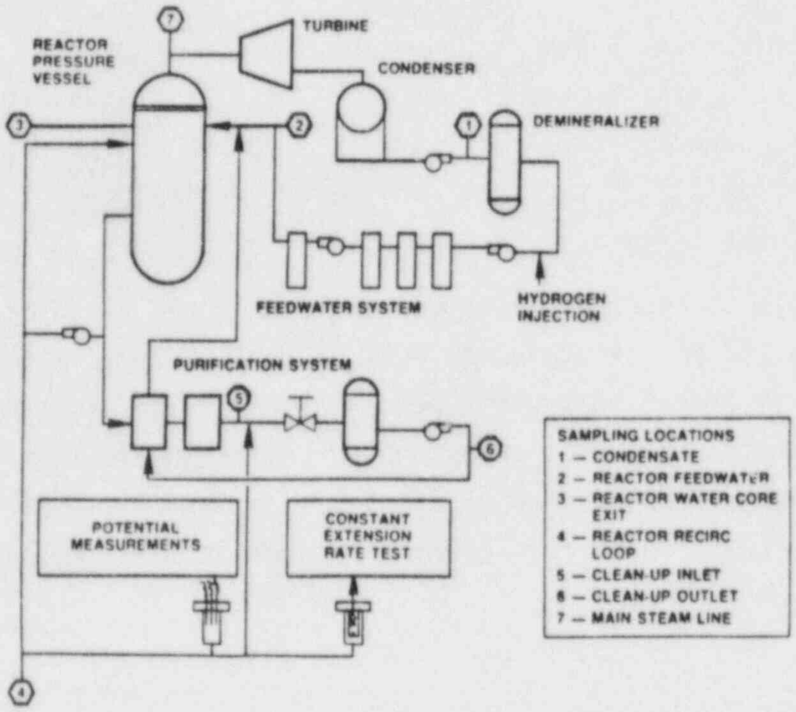

Figure 2. Hydrogen flow sheet for Dresden-2

This panel varies the hydrogen flow in proportion to feedwater flow to give a constant but adjustable hydrogen concentration in the reactor feedwater. Maintenance of feedwater hydrogen at 1.5 ppm results in dissolved oxygen concentrations of 20 ppb or less throughout the primary coolant system.

Oxygen from a liquid oxygen storage trailer provides 60 psi oxygen gas to add into the off-gas system at the first stage steam jet air ejector (SJAE). Oxygen flow is controlled manually based on the reading of oxygen meters sampling the recombiner outlet gas. A steady-state oxygen flow is set to maintain the oxygen concentration at the recombiner outlet between 8 and 12 volume %. When hydrogen injection rate increases are to be made, oxygen flow is increased first to be sure there is always excess oxygen.

The hydrogen addition system is not operated continuously. It is shut off for a variety of reasons, including maintenance activities on the hydrogen addition system itself, maintenance in other areas of the plant where there are high N-16 radiation levels due to hydrogen addition, and during the extinguishing of fires in the off-gas treatment system. In addition, during reactor startups and shutdowns, hydrogen addition is not used below 220 MW (e) (i.e., 25% power).

When the injection system is operating, it significantly reduces the oxygen concentration in the reactor water. Overall, the injection system availability has been about 87%. During 1984 the injection system was available for about 90% of the time and the oxygen was controlled below 20 ppb for 86% of the time (Figure 3).

It is worthwhile to investigate the nature of the time the reactor water was greater than 20 ppb oxygen because of the existence of a corrosion potential "memory effect" discussed later in the paper. As a result of this effect, the corrosion potential stays below the critical value for IGSCC for at least 10 hours after shutdown of the hydrogen injection system. The statistical data base shows that greater than 70% of the cumulative time the

Figure 3. Plot of all oxygen concentration data for calendar year 1984

reactor water was above 20 ppb was for periods of less than 10 hours in any given 24-hour period. This implies that over 93% of the total cumulative operating time since startup of Dresden-2 in March 1983 has been in the corrosion potential region that is "immune to IGSCC" for conductivity levels ≤ 0.3 μmho/cm.

The major problem causing shutdown of the H_2WC system has been off-gas fires. An extensive evaluation of the reasons for the off-gas fires revealed they were not directly related to H_2WC, but, rather, to the design of the off-gas piping system. The probable ignition source is migrated catalyst in the off-gas trains, and flow or pressure changes cause ignition by fluffing of the catalyst. Since the fires are located in the SJAE, after the condenser, the recommendations for eliminating the fires include elimination of a bypass line in the A-B off-gas trains (which is unique to the Dresden and Quad Cities Units), followed by poisoning or removal of the migrated catalyst in the remaining lines. Also, a change in the oxygen addition point to the steam diluted downstream region is being considered.

Plant Water Chemistry

Control of ionic impurities has been particularly good in Dresden-2 for the first 18-month cycle of H_2WC. Conductivity has been maintained at less than 0.2 μmho/cm for 98% of the time (Figure 4), and pH has been maintained in the 7 to 7.5 range. Ion chromatography studies of the coolant at Dresden-2 have shown that under continuous operation conditions, no more than 30 ppb carbonate, 5 ppb chloride and 5 ppb sulfate were present. The balance of the conductivity is made up of hydronium and hydroxyl ions. Thus, the species making up the conductivity at Dresden-2 are less aggressive than the sodium sulfate used to assess conductivity effects in the laboratory studies (7).

In addition to monitoring conductivity, pH, and oxygen, the H_2WC verification program also includes the measurement of various metals and isotopes in the feedwater and reactor water. Soluble and insoluble corrosion products have been collected continuously, with samples changed at roughly three-day intervals. Each sample fraction has been analyzed at Vallecitos Nuclear for Fe, Cu, Ni, Co, Zn, Cr, and Mn.

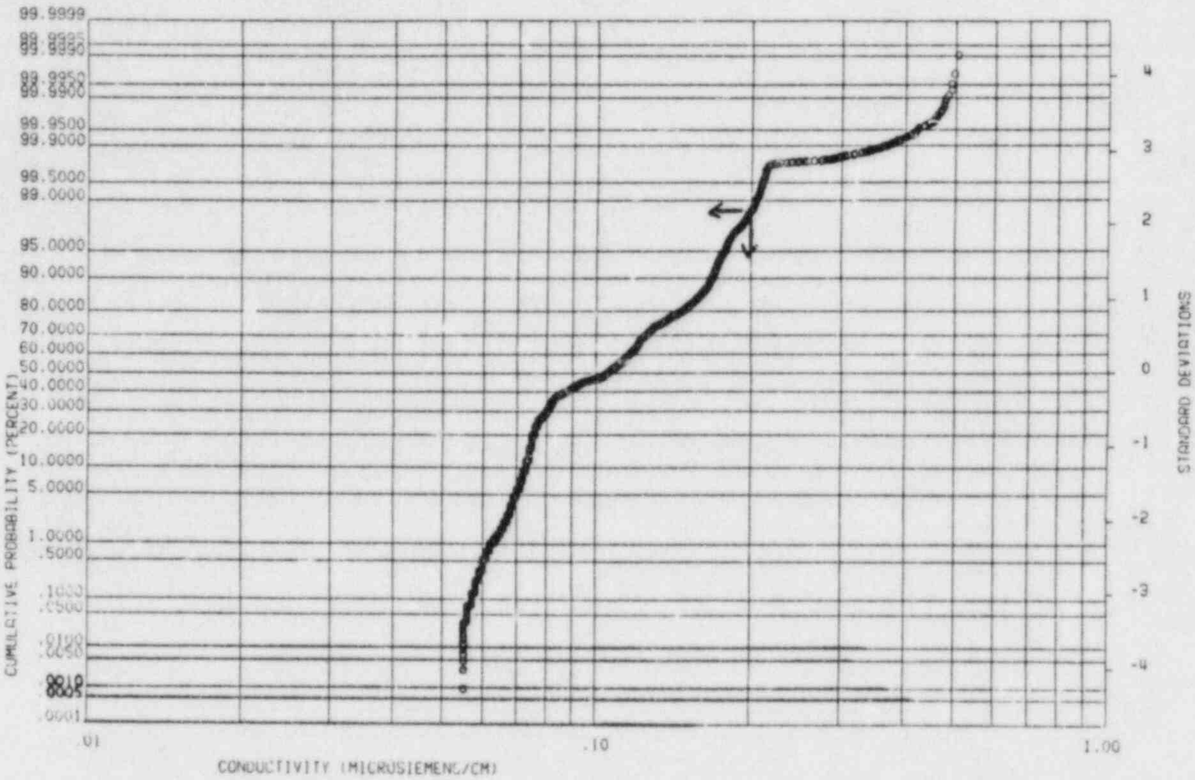

Figure 4. Cumulative Data on Reactor Water Conductivity for H_2WC Cycle 9.

The dominant impurity in the Dresden-2 feedwater is insoluble iron. Figure 5 shows the concentration of insoluble Fe as a function of time for a period since July 15, 1983. The 20-ppb spikes are the result of several condensate demineralizer changeouts over a short interval; the 40-ppb spike encompasses an orderly shutdown for pipe crack inspection and the restartup of the reactor. No long-term adverse consequences of hydrogen addition are evident. The time-base plots for other elements that were analyzed all show similar patterns, with no element showing an upward trend with time. The concentration of the other insoluble metals is generally less that 0.1 ppb. In the reactor water, the dominant activation product is Co-60, but no clear trend, either upward or downward, was observed. Values between 0.1 and 0.2 μCi/l are consistent with data from other BWRs.

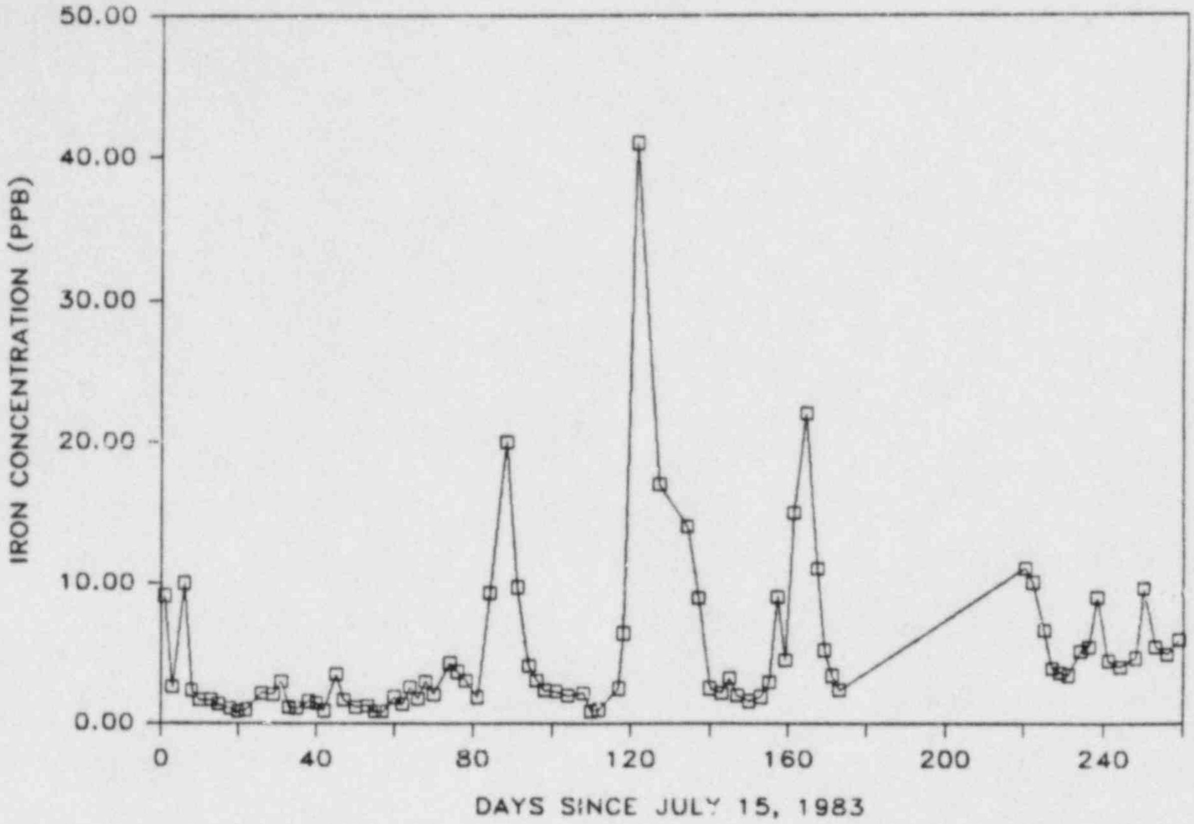

Figure 5. Feedwater insoluble iron measured during the H_2WC cycle 9 at Dresden-2.

Of the soluble species that were monitored in the feedwater, cobalt is the element of greatest concern because of its activation to Co-60 in the reactor core and accompanying potential for radiation buildup. Figure 6 shows the concentration of soluble cobalt in the feedwater as a function of time. The downward and stable trend of cobalt is readily apparent. This trend pattern is also observed for the other elements that were analyzed. In contrast, there was no definitive upward or downward trend in the cobalt data in the reactor water; average values of 40 ppt for soluble Co agree with measurements from other BWRs. Because of the operational practices changes to the condensate treatment system that were implemented during Cycle 9, it cannot be explicitly concluded that these trends are a result of hydrogen water chemistry. Nonetheless, there appear to be no adverse consequences of hydrogen water chemistry for soluble species in either the reactor water or feedwater.

Summarizing the corrosion product data, we have not seen any indication that hydrogen water chemistry has caused any detrimental change in the soluble

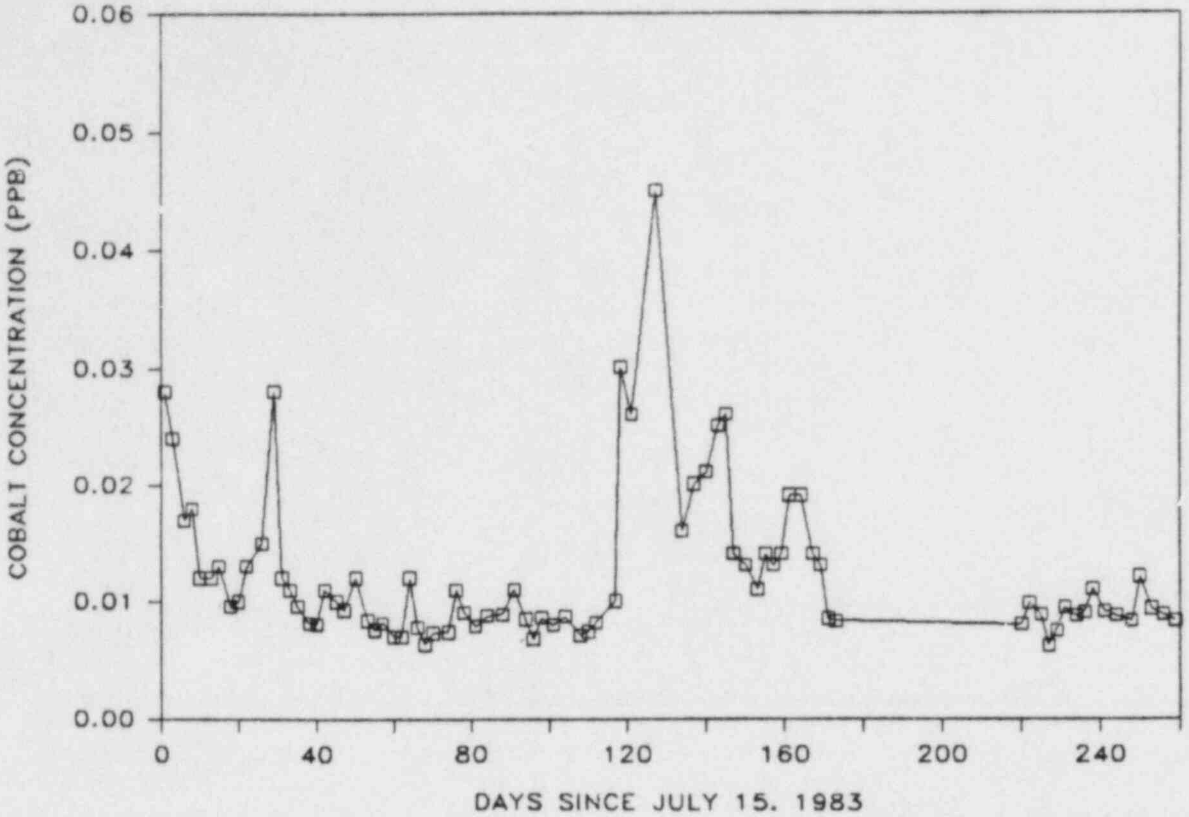

Figure 6. Feedwater soluble cobalt measured during the H_2WC cycle 9 at Dresden-2.

and insoluble corrosion product transport in the feedwater or the reactor water. In the feedwater, insoluble spikes are common to all BWRs, and clearly will propagate to the reactor water. The downward trend in feedwater solubles we feel is the result of a gradual implementation of good operational practices centered around the management of the condensate treatment system. The concentrations of the impurities in the feedwater and reactor water are consistent with data from other deep bed plants.

Radiation Levels

An important concern associated with H_2WC is the impact on radiation levels in the turbine building, plant environs and off-site. The earlier 30-day test had indicated an increase on the order of a factor of 4 to 5 in N-16 in the steam. Detailed measurements have now been made in and around the plant, indicating that although the impact of H_2WC is significant and measurable in the vicinity of the turbine building, it diminishes rapidly away from the plant and is negligible off-site. Figure 7 indicates the

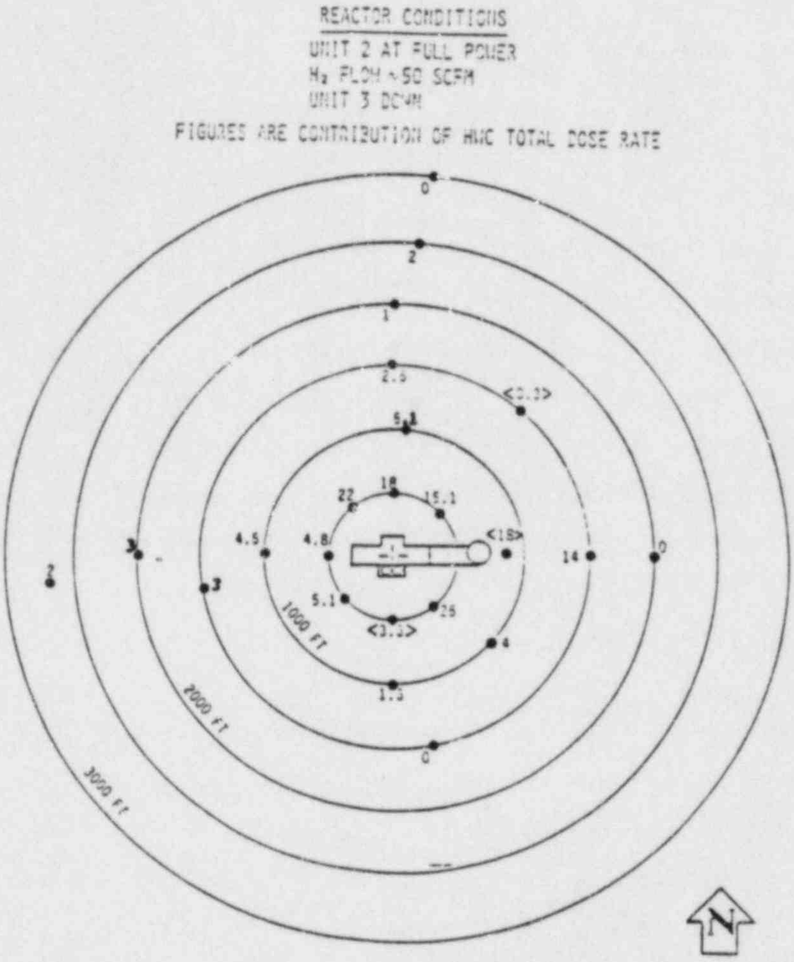

Figure 7. Dresden-2 Environs Dose Rate Map (μR/Hr)

contribution of H_2WC to the total dose rate on a series of contours drawn from the turbine building. Addition of Hydrogen Water Chemistry at Dresden-2 has been found to add about 5 man rem per year to the 24 man rem/year caused by nuclear fuel cycle activities (i.e., 29 man rem/year total to a plant population of about 1,000 people). However, in terms of exposure for IGSCC-related repairs, HWC is ALARA effective. For example, pipe replacement programs typically lead to exposure of 1500 to 2500 man rem. It must be noted, that the Dresden-2 plant layout and site location are favorable for reduced radiological effects of the H_2WC system. Other plants might not be so favorably constructed or situated and a current task in the EPRI program is evaluating the sensitivity of other plants to the radiological consequences of H_2WC.

Fuel and Plant Materials Characterization

Slow strain rate, crack growth, and constant load tests on a variety of plant structural materials are being undertaken in-plant to supplement the extensive laboratory data already available (7). These in-reactor tests are conducted in special autoclaves using reactor water fed from a header in the recirculation system. The results of stress corrosion and electrochemical potential studies provide strong evidence that stainless steel stress corrosion cracking activity at Dresden-2 has been arrested. A series of in situ stress corrosion tests were conducted on furnace sensitized stainless steel with the tests ongoing over about 2000 hours. No cracking was observed in either smooth or IGSCC precracked specimens tested while the plant was operated with reactor water at 20 ppb O_2 or less and reactor water conductivity at 0.3 μmho/cm or less. Electrochemical potential measurements on stainless steel made at the plant for over 3000 hours revealed potentials well below -350 mV (SHE) over 95% of the time. This potential is considered to be a threshold for IGSCC for the conductivity values listed above. Coolant oxygen content excursions up to 200 ppb for limited periods of time for up to about 15 hours during the constant extension rate tests did not result in IGSCC. Electrochemical potentials during these short-term excursions did not increase to the values observed under normal operation at 200 pbb oxygen indicating the presence of a memory effect (Figure 8).

Perhaps the most revealing test result is the constant extension rate test conducted on a furnace sensitized and laboratory IGSCC precracked specimen that had seven hours of in-plant test time with 200 ppb oxygen. The balance of the in-plant test time was within the H_2WC regime. No IGSCC in addition to that in the precrack phase was noted. This result is consistent with in-service inspection observations. The recirculation piping was inspected in November 1983 in response to an NRC directive since IGSCC flaws had been detected earlier in a large-diameter line. The inspections revealed no further growth of the existing flaws. Further inspections are planned in October 1984 during the fuel outage.

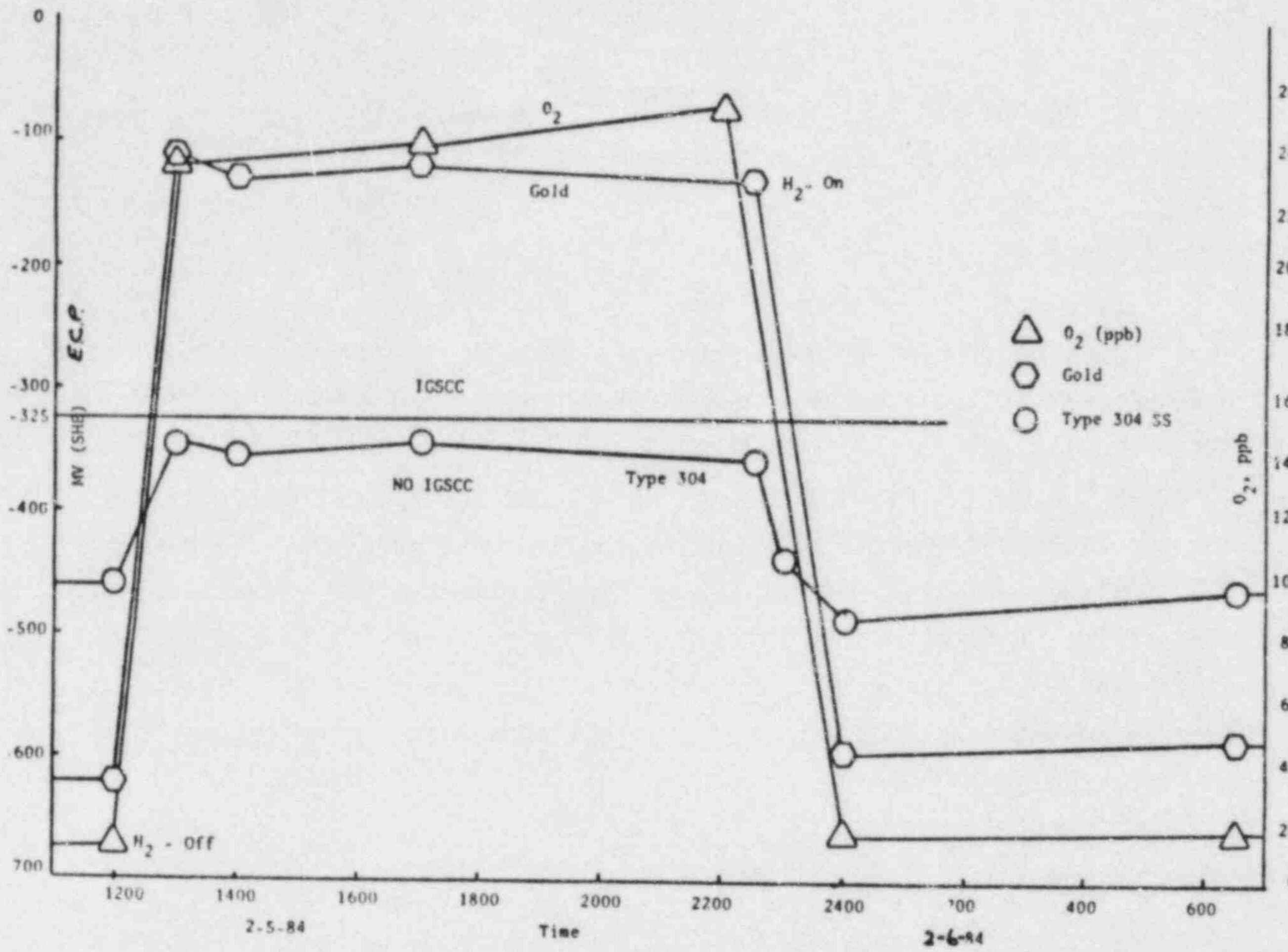

Figure 8. Example of electrochemical potential (ECP) memory effect during H_2WC operations.

Additional in-plant and laboratory studies have shown that stress corrosion and or corrosion fatigue of the other major structural alloys including Inconel 600, carbon steel, and low alloy steel are effectively eliminated by application of H_2WC. No effects were found for martensitic stainless steel. Some increase in carbon steel general corrosion rates can be expected, but resultant corrosion is well within design tolerances (7).

The concern over the long-term behavior of fuel, particularly with respect to hydrogen pickup and hydriding, prompted a plan for more extensive examination of fuel rods and channels. For this purpose, precharacterized fuel assemblies, both prior irradiated and unirradiated, were inserted into Dresden-2 during the last outage. At the end of this cycle, they will be non-destructively examined, and selected rods will be removed from the assemblies, transported to the hot cells, and destructively examined to determine hydrogen pickup and hydride morphology.

Future Plans

Based on experience to date, several areas for improvement of the hydrogen water chemistry system at Dresden-2 have been identified. To reduce costs, the plant will convert to liquid hydrogen storage and supply for the next cycle. Also, better system control will be achieved through control room panel modifications to bring the oxygen and hydrogen supply monitors to a single observation point so that any changes to the hydrogen injection rate can be rapidly reflected by changes to the oxygen supply to the SJAE. In addition, all H_2WC data will be recorded by the process computer so that an integrated printout of plant operations including hydrogen water chemistry parameters can be produced.

Finally, another development which we believe will be important in verifying the mitigating effects of H_2WC on IGSCC is the installation of a continuous UT monitor on the 28" flawed pipe mentioned earlier. Commonwealth Edison is installing a high-temperature UT sensor package developed by Amdata Systems under EPRI funding during the upcoming 1984 fuel outage.

CONCLUSIONS

To date, the results of the combined laboratory development program and in-plant verification of hydrogen water chemistry are very encouraging, and indicate that an alternative BWR water chemistry which will mitigate IGSCC in stainless steel recirculation piping during power operation can be specified. Experience shows that the maintenance of ≤20 ppb oxygen by continuous addition of hydrogen and ≤0.2 μmho/cm conductivity, through efficient operation of condensate and reactor water demineralizers, can be achieved in a commercial power plant for >90% of the time at power with minimum system impact. The off-gas fires are not attributable to H_2WC but, rather, may be aggravated by the unique off-gas piping system at Dresden-2 and the engineering solution appears to be straightforward. Radiation level increases due to H_2WC, although not a concern at Dresden-2, could pose a more significant problem at other plants. This must be evaluated, and recommendations on shielding requirements, etc., provided to other BWR owners.

No serious detrimental effect of H_2WC on the performance of BWR structural materials has been identified in short-term laboratory and in-reactor tests. Longer term in-plant tests are in progress to confirm that H_2WC is generally a less aggressive environment than normal oxygenated BWR water.

Finally, the estimated cost for the installation of H_2WC is about \$3 million, with subsequent operation costing from \$300K to \$500K per year. The operating costs are strongly influenced by the cost of hydrogen. It appears that for many plants the use of liquid hydrogen as a delivery/storage system would be optimum. Installation of a hydrogen recovery/recycle system may also be cost-effective at some plants. In some cases lower hydrogen costs may be achievable through the use of advanced electrolytic generation on site. These options are covered in comprehensive detail in a recently published EPRI report (8).

ACKNOWLEDGMENTS

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R&M PRACTICES IN COMMERCIAL AVIATION

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Introduction

During 1982-83, Los Alamos Technical Associates (LATA) and American Management Systems (AMS) conducted an extensive review of design and operations practices in the commercial air transport industry focused on reliability and maintenance for the Electric Power Research Institute (EPRI).[1] The genesis of this review was a discussion of the technical and environmental similarities of the commercial air transport and commercial nuclear power communities and the potential opportunities for experience transfer from air transport to nuclear power. This paper will focus on those similarities believed to be of particular interest to the nuclear power community. Figure 1 summarizes some interesting characteristics of the air transport community. Figure 2 identifies some interesting differences.

The Review Process

The review team consisted of Mr. A.M. Smith, an engineer with extensive nuclear systems reliability experience and two retired airline vice presidents, Mr. John McDonald (who has also had extensive air transport manufacturing experience) and myself. Our external sources included two large air transport manufacturers, a large NSSS manufacturer, a large nuclear plant architect/engineer, an operating nuclear power plant, and three large airlines. Most of the external information was acquired by on-site visits at these activities.

[1] EPRI Research Project TPS 82-67

COMMERCIAL AVIATION - SOME KEY CHARACTERISTICS

- HI - TECHNOLOGY CONTENT
  - e.g. COMPOSITE MATERIALS, MICROPROCESSORS, PRECISE CONTROL SYSTEMS, CYCLIC STRESS IN LONG LIFE STRUCTURES.

- HIGH SAFETY RISKS
  - 5,000,000 TAKE-OFFS PER YEAR
  - 2,800 IN-SERVICE AIRCRAFT

- AFFECTS LARGE POPULATION
  - 300,000,000 PASSENGERS PER YEAR

- CRITICAL MAN-MACHINE INTERFACE
  - 37,000 PILOTS
  - 45,000 MECHANICS

- INTENSIVE REGULATION
  - 400 FAA OPERATIONS INSPECTORS
  - 280 FAA MAINTENANCE INSPECTORS

- LONG TERM OPERATION
  - DESIGN FOR 20 YEAR ECONOMIC LIFE (707/DC-8 NOW 25 YEARS OLD)

- CAPITAL INTENSIVE
  - AIRCRAFT INVESTMENT ALONE > 22 BILLION

DOESN'T THIS SOUND LIKE THE NUCLEAR INDUSTRY?

FIGURE 1

DIFFERENCES ALSO EXIST

FOR EXAMPLE

- INDUSTRY STRUCTURE
  - DESIGNER/BUILDER VS OPERATOR
  - IN-SERVICE SUPPORT
- REGULATION
  - SAFETY PLUS PROMOTION
  - NO PUC'S
  - NO PUBLIC HEARINGS
  - USE INDUSTRY PERSONNEL (DER/DMIR)
  - REQUIREMENTS ORIENTED (NOT PRESCRIPTIVE)
- SIZE AND NUMBER OF OPERATING UNITS
  - SMALL SIZE/LARGE QUANTITY VS LARGE SIZE/SMALL QUANTITY
  - BUT UTILITY VS AIRLINE CAPITAL INVESTMENT IS SIMILAR
- CONTINUITY OF OPERATION
  - CONTINUOUS VS CONTINUAL
  - MAINTENANCE ACCESS
- INDUSTRY MATURITY
  - 50 VS 25 YEARS
  - A GOOD REASON TO LOOK CLOSELY

FIGURE 2

ROLES IN THE AIR TRANSPORT COMMUNITY

Research

The principal scientific research supporting air transportation has historically been conducted by NASA and its predecessor, NACA. Primary technologies have been also dynamics, propulsion and structures.

Engineering research, particularly related to aircraft safety, has historically been conducted by the FAA and its predecessor, CAA.

Engineering research, particularly related to specific design development, has historically been conducted by manufacturers.

Engineering research, particularly in-service testing of prototype designs of systems and equipments, has historically been conducted by the larger air carriers.

Regulation

The Federal Aviation Act of 1958 assigns the FAA responsibility for insuring safety and promoting air commerce. The Federal Aviation Regulations cover both design and operations.

Compliance with the myriad of design requirements is regulated with the aid of a number of senior technical specialists who are employees of the manufacturer <u>and</u> also FAA Designees. These persons provide a valuable adjunct to the FAA in assuring regulatory compliance while achieving the objectives of the Project Engineer in a range of technologies that if covered directly by FAA employees would severely constrain the design and construction process. The approval of the Type Certificate is made by a FAA Type Certification Board which monitors the design with the aid of the on-site FAA and FAA Designee personnel.

The principal underlying philosophy of air transport design is that single failures shall not threaten safety, and detail design shall inhibit maintenance errors. Compliance with the air transport operations requirements, including maintenance, is regulated by the FAA by Operations and Maintenance Inspectors assigned to each air carrier. Maintenance Inspectors are responsible for approving each air carriers maintenance program and for continuous surveillance of its performance. The initial preventive maintenance program for each new aircraft placed in service (developed by the initial users with the assistance of the airframe, engine and other component manufacturers) is approved by a Maintenance Review Board which includes representative maintenance inspectors.

Design/Manufacture

The end item, a transport aircraft with an FAA Type Certificate, is the sole responsibility of the airframe manufacturer. The propulsion system manufacturer is the major subcontractor. He and all other system contractors and fabricators respond to the airframe manufacturer's design, performance and interface specifications. Except where airframe manufacturer proprietary designs are involved, purchases of spare parts, and, in many cases, independent negotiations of warranties and guarantees are common practice by the larger purchasers.

Operations

The end item,, a transport aircraft complying with all of the terms of the purchaser's agreement with the airframe manufacturer, is placed in service only after approval of the required changes/amendments to the air carrier's Operating Certificate. These include the initial preventive maintenance program, which provides for preventive maintenance for the entire life cycle fo the entire aircraft. The working relationship between the Maintenance Inspectors and the air carriers are usually defined in detail by an individually developed "Reliability Program". This program specifies the airline's internal quality controls and information systems, the processes used to modify its maintenance programs, and the procedures for maintaining records and preparing reports.

RELIABILITY AND MAINTAINABILITY IN DESIGN

Four key findings were related to design:

- Project/system integration. The airframe manufacturer is a single focal point of responsibility for overall project/system management and integration.

- Reliability and maintainability program. The airframe manufacturers have found a formal R&M program to be vital to minimizing safety and economic risks. R&M requirements have been built into the specifications system on an equal basis with performance requirements. (See Figure 3).

- Quantitative R&M goals. End item quantitative R&M goals and allocation of these goals to systems and structures have been an integral part of the design process being viewed as absolutely necessary to market new designs in a highly competitive marketplace, to support guarantees and warranties and to communicate end item objectives to both internal and external resources responsible for detail design. (See Figure 4).

- Guarantees and warranties. The airlines have always expected and received a high level of product support. Aircraft manufacturers are proud of their record of support of their products over their useful life irrespective of changes in their location or ownership. Essential elements of this support (technical data and supply) are provided in accordance with Air Transport Association specifications. During the 60s and 70s, additional product support commitments in the form of significant guarantees and warranties have escalated in both quality and scope. Two basic benefits are now almost universally provided:

    - Correction of design deficiencies through modifications at no cost for parts or labor.

EVOLUTION OF AIRCRAFT MANUFACTURERS' RELIABILITY AND MAINTAINABILITY PROGRAM

R&M Program Elements	DC-8	DC-9	DC-10
Load and Stress Analysis	✓	✓	✓
Development Testing	✓	✓	✓
Material Standardization and Q.C.	✓	✓	✓
Failure Reporting, Corrective Action, Product Improvement	✓	✓	✓
Apportioned R Goal with Running Predictions		✓	✓
System Simplification Program		✓	✓
Component Selection Program		✓	✓
Minimum Equipment List		✓	✓
Failure Mode & Effects Analysis		✓	✓
R Operational Simulation Model			✓
Fault Isolation Goal and BITE			✓
Design "Input" Meetings			✓
Supplier R Guarantees			✓
Supplier Design Reviews			✓
Failure Predictability in Designs			✓

IATA

FIGURE 3

KEY FINDING III

QUANTITATIVE GOALS

- USE OF VERIFIABLE QUANTITATIVE R&M GOALS IS A STANDARD PRACTICE
  - MANUFACTURER BUILDS INTO HIS SPECIFICATIONS
- MANUFACTURERS MOTIVATED BY
  - COMPETITION
  - INCREASING IMPACT OF UNRELIABILITY
  - INCREASING IMPACT OF OPERATOR MAINTENANCE EXPENSE
  - GUARANTEES AND WARRANTIES (TIED TO GOALS)
  - COMMUNICATION WITH 600 PERSON DESIGN TEAM + SUPPLIERS
- RECENT TREND TO SATISFY A SPECIFIC FAA SAFETY GOAL DURING TYPE CERTIFICATION

LATA

FIGURE 4

- Provision of additional spare assemblies on loan to fill logistic support pipelines until guaranteed reliabilities are achieved.

These programs may require specific actions by users such as return of failed hardware and supplying failure statistics. These programs are usually the subject of extensive negotiation. (See Figure 5).

- Regulation. Regional FAA offices manage regulatory activities affecting specific manufacturers, including design reviews, testing, manufacturing inspection, engine and aircraft type certification, maintenance review boards, etc. They also act as technical liaison with manufacturers in negotiations affecting Airworthiness Directives.

  The employee/FAA Designees provide a highly effective means for state-of-the-art application of regulations affecting design.

KEY FINDING IV

GUARANTEES AND WARRANTIES

- AIRLINES AND MANUFACTURERS ACUTELY SENSITIVE TO PRODUCT SUPPORT NEEDS
- COMPETITION AND ECONOMICS CREATED "QUANTITATIVE" SUPPORT REQUIREMENTS--i.e., GUARANTEES AND WARRANTIES
- AIRLINES NOW DEMAND BASICALLY TWO FORMS OF SUPPORT PROTECTION:
  - CORRECT DESIGN DEFICIENCIES AND PROVIDE RETROFITS COST-FREE (GUARANTEES)
  - PROVIDE SPARES COST-FREE IF PRE-ESTABLISHED UNRELIABILITY LIMITS ARE SURPASSED (WARRANTIES)
- AGREEMENTS OFTEN INCLUDE RECIPROCAL CLAUSES, e.g.
  - RETURN FAILED HARDWARE
  - COLLECT DATA TO SUPPORT PROGRAM
- REMEDIES RECOGNIZE SUPPLIERS RESOURCE LIMITS

IATA

FIGURE 5

RELIABILITY AND MAINTENANCE IN OPERATIONS

Three key findings were related to operations:

- Preventive maintenance. Effective PM programs are a hallmark of the air transportation industry. The methodology now called Reliability-Centered Maintenance reflects the intensive effort by this community to eschew intuition and industrial folklore in favor of logical processes for defining PM requirements. The result has been:

  - Significant reductions in equipments subject to scheduled overhaul.

  - Significant extension of the intervals between overhaul.

  - Virtua'ly no increase in maintenance costs over the past 20 years, (see Figure 6), during which their safety record has steadıly improved.

- In-service reliability management. Each major airline has an active reliability control program developed in accordance with guidelines published by the FAA.[2] These programs monitor reliability of aircaft, by type, as a whole and of each of their critical systems, giving guidance to management in allocating corrective resources. The resultant actions may be changes in procedures, standards or design. A wide range of techniques are used, some focusing on meeting quantitative standards, others on ranking techniques. This body of processes grew out of joint efforts by an FAA/industry study group in the early 60s. This work, together with RCM, represents the zenith of FAA/airline efforts to find better ways to manage in-service reliability.

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[2] FAA Advisory Circular 120-17A

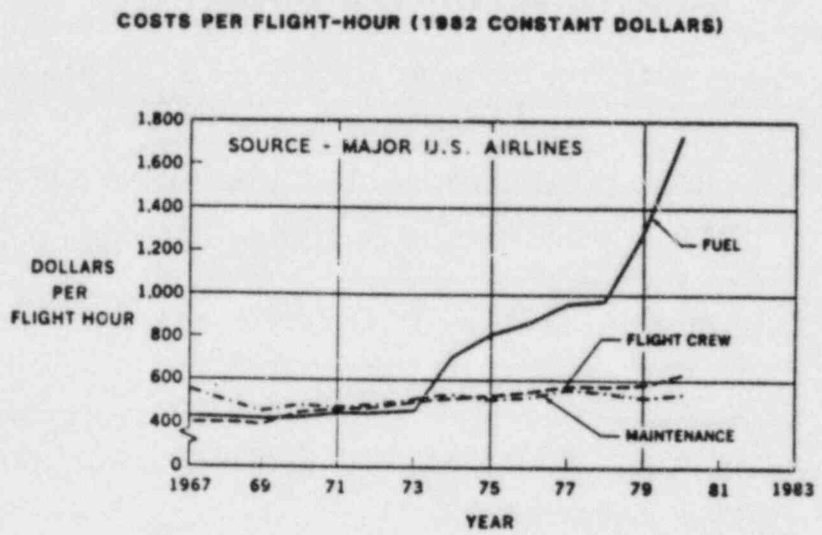
COSTS PER FLIGHT-HOUR (1982 CONSTANT DOLLARS)
SOURCE - MAJOR U.S. AIRLINES
DOLLARS PER FLIGHT HOUR
1,800
1,600
1,400
1,200
1,000
800
600
400
0
FUEL
FLIGHT CREW
MAINTENANCE
1967
69
71
73
75
77
79
81
1983
YEAR

FIGURE 6

- Information systems. The airlines have individually funded extensive efforts to design information systems that specifically support maintenance management. These provide:

    - Feedback for product improvement
    - Feedback for new aircraft designs
    - Feedback for PM program changes
    - Feedback for change in design regulations
    - Operation of reliability control programs
    - Substantiation of guarantee and warranty claims
    - Feedback to logistic support systems
    - Initiation of safety-critical alerts

    They focused on two different categories of data:

    - Events which are of sufficient importance to require individual management action.

    - Statistics which cumulate events that are important because of their quantity and therefore their economic impact.

- Regulation. On-site regulation of air transport maintenance is provided by resident FAA Maintenance and Avionics Inspectors. Their work includes:

    - Visual observations
    - Documentation audits
    - First level review of changes in processes or design requiring FAA approval
    - Review of all reports required by the Federal Aviation regulations and monitoring of follow-up actions

    Higher level reviews are conducted at regional and headquarters levels.

The Air Transport Association coordinates industry level interaction with the FAA with particular emphasis on Airworthiness Directive releases and regulatory changes.

Summary

The air transport community has developed highly effective means for managing reliability and maintenance in an environment requiring the highest degree of safety. Their success may point the way for approaching similar problems in activities, such as nuclear power generation, where public safety is of paramount interest.

SOURCE TERM RESEARCH AND PROGNOSIS

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ABSTRACT

The work sponsored by EPRI on source term technology is discussed (source terms describe the fission product releases to the environment in a severe hypothetical accident). The experimental programs include (1) fission product release from fuel, (2) fission product transport in the reactor primary circuit, (3) aerosol behavior in reactor containment, (4) aerosol scrubbing by water pools, and (5) hydrogen combustion in the containment. Code development work is also included.

1.0 INTRODUCTION

The first nuclear reactor was built under the West Stands at the University of Chicago. Its designers, well aware of the potential dangers, provided crude but effective shutdown measures to be used in the event of an accident. For example, they stationed a man with an axe to cut the rope holding a shutdown rod and assigned another man to break a large bottle of boric acid placed on top of the pile.

In that early time, the approach to reactor safety was obviously less sophisticated than now; however, we now have over-reacted in the prediction of the consequences of an accident. Thus there is concern that the use of so-called "conservative" or erroneously large predictions of the consequences of low-probability reactor accidents may lead to a counterproductive over-reaction in emergency response planning and certainly an unnecessarily negative public perception of the potential outcome of a reactor accident. It is now the objective in the reactor safety community, both the governmental and the industrial components, to do "realistic" accident evaluations as contrasted to the past "conservative" evaluations.

EPRI, along with several other research groups, has been working hard to establish the methodology and a data base to present a credible case for lower source terms. This paper will describe the EPRI efforts.

2.0 ANALYSIS FOR SURRY

In order to become proficient in using the consequence analysis methodology EPRI has been carrying out work at Science Applications, Inc. to determine the risk due to a hypothetical degraded core accident at Surry. This is a large dry containment PWR and was analyzed in WASH-1400. It was of interest to see how more recent codes and data changed the WASH-1400 results. The analytical methods used in the study mostly consist of the NRC group of computer codes. These were MARCH-2 (general thermalhydraulics), the EPRI code PSTAC (primary system thermalhydraulics), TRAP-MELT-2 (primary system fission product transport), MATADOR (containment fission product transport), CORCON (core-concrete interaction), and CRAC-2 (public risk evaluation). Results of recent experimental programs were used to specify in-vessel and ex-vessel source material release rates and to define best-estimate decontamination factors for water pool scrubbing in applicable accident sequences. Some of these experimental programs will be described later in the paper.

The specific accident sequences being analyzed were determined from a re-examination of the classification of dominant accident sequences for the Surry PWR as given in WASH-1400. This exercise has lead us to conclude that S_2C (small LOCA with containment injection spray failure) is not a likely key sequence and left TMLB' (transient with failure to recover electric power) and V (interfacing LOCA) as the risk dominant seuqences requiring re-analysis. Thermalhydraulic calculations for the TMLB' sequence, in combination with results of structural capability studies, predict that the Surry containment would remain intact for at least the first 12 hours of the accident. This contrasts with the WASH-1400 prediction of catastrophic failure by overpressure in less than four hours. The resulting revised source term for TMLB' is several orders of magnitude lower than the WASH-1400 estimate. Realistic modeling of plant systems and geometry for the V sequence indicate continuous flooding of the pipe break in the Safeguards Building at Surry throughout the periods of principle radionuclide release. Plateout of fission products on system surfaces along the transport path and retention by the water pool in the Safeguards Building are expected to significantly reduce radioactivity escape to the atmosphere. Preliminary estimates have produced a source term which is considerably lower than the WASH-1400 value assigned to this sequence.

The risk to the public, represented by the revised source terms, is being assessed with improved versions of the computer codes compared to that used in the WASH-1400 work. The site population and meteorological data used in the calculations also come from the WASH-1400 files. The results of the preliminary calculations indicate very few or no early fatalities for the revised source terms, and the latent fatality (cancer) risk is approximately one to two orders of magnitude lower than the corresponding WASH-1400 results.

In the process of using the codes a number of problems surfaced. Some of these problems have very likely been corrected in later code versions since the work was done. Some of these problems were:

- In March 2, there appears to be an overly simple treatment of heat transfer in the PCS - particularly for the steam generator.

- March 2 gave an averaged core exit temperature which was in error for higher core melt fractions.

- For March-2 the maximum time step permitted during the calculation of core uncovery can have a significant effect on the predicted core heat up rates.

- CORCON-Mod 1 is inapplicable when the dense layer solidifies.

- The lack of coupling between CORCON-Mol 1 and March-2 caused difficulty.

- The welt-mixed assumption in TRAP-MELT-82 is not appropriate for long pipe runs as in the "V" sequence.

- TRAP-MELT-82 lacks simultaneous analysis of heat and mass transfer. Also the heating effects due to fission product decay are not included.

- MATADOR is difficult to use because a large effort is required to assemble input.

- MATADOR does not include aerosol particle growth due to steam condensation.

- The diffusiophoretic particle deposition model in MATADOR gives erroneous rates when applied to steam superheated conditions.

- When calculations beyond 12 hours were attempted with CORCOM-Mod 1 the calculations became unstable.

- PSTAC (EPRI code) does not treat steam condensation along the PCS pathway.

- PSTAC does not consider the effects of aerosol transport and aerosol deposition on heat transfer and fluid flow.

3.0 PHENOMENA INVOLVING THE DEGRADED CORE

The next several sections of this paper will be devoted to describing the EPRI program directed at establishing a data base which can be used for improved evaluations of degraded core accidents.

Core overheating caused by a sustained undercooling condition, is a serious manifestation of a degraded core accident. Section 3.1 will describe experiments underway to clarify various aspects of fission product release from the fuel under these conditions. Section 3.2 will describe experiments on debris bed cooling.

3.1 FISSION PRODUCT RELEASE FROM LWR FUEL

EPRI is sponsoring a project at Argonne National Laboratory to determine the chemical form and rates of the fission product releases. In this work, a section of a radioactive fuel element is heated in a vacuum, and the gaseous species that escape are identified and measured using a mass spectrometer.

In early results using non-radioactive fuel simulants, at temperatures below 1300°C, only Cs and CsI were observed. Around that temperature stannous telluride (SnTe) was found, presumably because of chemical reactions between tin from the Zircaloy-cladding and Tellurium. At higher temperatures, around 1600°C, two more gaseous species were observed, CsTe and I. The former is believed to result from a reaction of gaseous, cesium with a $ZrTe_2$ layer on the inner surface of the fuel, the latter is probably due to thermal decomposition of CsI. Studies with radioactive fuel elements are now underway.

The work just described will give us valuable information on the chemical state of the fission products. In order to better evaluate the physical state of the fission products and to make the observations in a more prototypical fashion, in-pile experiments are also being carried out. These experiments are to be undertaken in the Argonne TREAT reactor (the STEP Program) and are to be done as soon as possible in order to make the data quickly available[1]. This work involves the support of DOE, NRC, Ontario Hydro and Belgonucleaire as well as EPRI.

The experimental system concept consists of a bundle of four pre-irradiated, Zircaloy-clad LWR fuel pins contained in a capsule-type vehicle made up of a cylindrical pressure vessel, a highly insulated, inert flow channel, a steam source and sink, and a source term measurement system. A relatively simple measurement system has been conceived that will provide chemical and physical characterization of selected fission products with spatial and temporal resolution by means of both real-time measurement and sequential sampling with post-test analysis. The characteristics of the reactor are such that a wide range of preselected time-energy sequences can be used. The part of the hypothetical degraded core accident to be simulated is from fuel exposure to steam to fission product release. During the course of the simulation, appreciable Zircaloy oxidation will occur. Emphasis will be on the measurement of the aerosol characteristics close to the fuel pins. Pre-irradiation of the fuel to about one-atom-percent burnup, will provide an adequate fission product inventory.

The first experiment, the STEP-1 AD (PWR large break severe accident) was conducted on June 19, 1984. The reactor transient was terminated via a planned scram after about 20 minutes. Test completion was achieved some 7 minutes later by terminating steam flow to the in-pile vehicle and opening the system pressure control value.

During the test, system temperatures behaved essentially as anticipated. Temperatures at the inner wall of the ZrO_2 flow channel reached 3200°F at fuel mid-height at about 12 minutes into the power transient, at which time the Type S (Pt/Pt-Rh) thermocouples failed. This is within 200°F of the melting point of Zr. Zircaloy oxidation began at about 7 minutes, as indicated by the H_2 monitor, and accelerated rapidly at 10 minutes. The gas inside the vehicle was essentially all H_2 from that point to the end of the test. Peak primary vessel temperature reached 1750°F about 5 minutes after reactor shutdown; this was 200-300°F higher than anticipated and occurred 5-10 minutes sooner than expected. This is thought to be due to an assumption in the pretest calculations of a too low insulator conductivity.

The temperature control, steam supply, and pressure control systems worked well throughout the test, although several pressure events occurred that are, as yet, only partially understood. These events may be characterized as pressure increased followed by relief. The most plausible explanation at this time appears to be accumulation of liquis cesium hydroxide on 20 μm mesh filters/flow balances which over a number of minutes builds up pressure as the liquid fills in the screen holes. Eventually a 20 psi pressure forces the liquid through the screen, relieving the pressure and the process of liquid accumulation on screen holes begins again. This phenomena is peculiar to the STEP design and is not expected to have affected the experimental simulation and associated data.

Post-test γ-spectra of the aerosol filters indicate the presence of Cs^{137}, Cs^{134} and I^{131}. Measurements made during the reactor transient have yet to be analyzed. The aerosol filter measurements, along with high radiation levels, primarily Cs^{137} and Cs^{134}, observed in primary system components inside the auxiliary box, provided assurance of significant fission product release from the test fuel. The presence of radioactivity in certain auxiliary box components was not anticipated, and the lack of shielding in that area made disassembly difficult. Portable shields were designed and constructed to reduce worker exposure to radiation during disassembly. Its presence there is thought to be related to the pressure events related above. No radioactivity escaped the primary system.

Initial hodoscope data indicate some early upward fuel motion near the top of the bundle that may be explained simply as axial thermal expansion, and some downward motion in the middle third of the bundle later in the test, characteristic of limited collapse after cladding disruption.

Impact on STEP-2 hardware and operations of the pressure events and radioactivity in auxiliary components is being evaluated. STEP-2 is a BWR TQUW simulation scheduled to be run in late October.

3.2 "DEBRIS BED COOLABILITY"

In the aftermath of a severe accident, after the core has melted, and lost its configuration, the question that may need to be answered is whether the resulting core debris bed can be adequately cooled by natural circulation of water and steam through the bed, or whether the bed will dry out and continue to overheat. If the debris can be adequately cooled, the accident can be terminated, and the containment will not be in danger of failing (e.g., TMI-2). Two small EPRI projects at UCLA and the University of California at Berkeley have been studying the coolability of a self-heating particle bed under pool boiling and forced-flow conditions and the quenching of such a hot debris bed by top or bottom flooding. This will help in developing experimental correlations and theoretical analyses of this type of scenario. Our current tentative conclusion from this work is that, given an adequate water supply, a debris bed made of large particles ($\sim$1cm) can be cooled either in or outside the vessel. When recovering from a severe accident, the debris bed is probably quenched better by bottom flooding rather than top flooding, because under some circumstances, the latter might not be as effective.

4.0 FISSION PRODUCT TRANSPORT

In this section, a number of projects will be described which involved fission product transport from the core through the primary coolant system into the containment and to the environment. Sections 4.1 covers code studies, whereas Sections 4.2 through 4.4 cover experimental work.

4.1 THE RAFT CODE

To predict the behavior of fission products after they leave the core region, a model would be useful that assesses the complex physical-chemical interactions of hot vapors, their condensation into aerosols, and their subsequent deposition and transport through the primary coolant system of a reactor. One such model is a computer code called RAFT (Reactor Aerosol Formation and Transport) which predicts the size distribution and composition of aerosols formed from condensation of volatile fission products and non-radioactive control rod materials released in postulated LWR accidents.

RAFT is presently being applied to many ongoing experimental and analytical programs, e.g., tests being conducted in the Marviken reactor, the TREAT (STEP) experiments being conducted at Argonne National Laboratory and large-scale aerosol containment experiments at the Hanford Engineering Development Laboratory. Significant results obtained in these simulations are summarized as follows:

- Parametric calculations have shown that gas pressure, concentration of fission product vapors and gas cooling rate are important variables.

- In the presence of control rod material vapors, Ag and In self-nucleate and provide sites for condensation of CsI and CsOH. In the absence of Ag and In vapors, CsI homogeneously nucleates with CsOH condensing on the CsI particles; but at high-mass loadings, the opposite occurs.

- The fission product vapors are removed through condensation on particles or on structural surfaces. Vapor condensation is sensitive to the surface temperature. For accidents in which the surface temperature remains below the dew point, a substantial fraction of fission product vapor is calculated to be removed via condensation on the reactor upper-plenum structure.

We hope to verify the above observations by validating RAFT with experimental data. Preliminary predictions using RAFT appear to indicate a rather large retention for fission products in a reactor's cooling system as compared to other computer codes.

4.2 LABORATORY TRANSPORT EXPERIMENTS

EPRI is involved also in acquiring experimental data on fission product transport in the primary coolant system in a degraded core accident.

At Argonne the transport of non-radioactive fission product stand-ins is being studied as they flow in a steam/hydrogen environment down a hot stainless steel duct. The duct is 12 feet long. The hot end is 1000°C and the temperature decreases to 500 C at the cold end. The experiments are being carried out at a somewhat elevated pressure of about four atmospheres. Mixtures of the chemical species involved will be used. It is felt that this may be of special significance since, for example, cesium hydroxide may remove oxide films from the stainless steel allowing more rapid reaction of the other fission product species, particularly tellurium with stainless steel. Several experiments have been carried out. Early observations indicate that cesium iodide aerosol particles are liquid. Experiments are about to start on cesium hydroxide and various mixtures as appropriate.

An additional set of experiments is underway which experimentally investigates the question of revolatilization of fission products due to fission product decay heating after they have been deposited in the primary system. In these experiments, mixtures of the deposited materials are placed in a stainless steel boat and heated in a flowing steam-hydrogen atmosphere to a pre-selected temperature (i.e., 500 C to 1000 C). The volatilized material and its retention in the boat has been measured. The picture from these experiments which is emerging is that cesium iodide at the higher temperatures (i.e., 1000°C) can revolatilize while cesium hydroxide and tellurium at, for example, 750°C undergo decreased volatility by complicated mechanisms which need further investigation.

4.3 MARVIKEN AEROSOL TRANSPORT EXPERIMENTS

EPRI and the NRC are working with eight countries in an approximately <u>full-scale</u> series of tests at Marviken, Sweden to study the attenuation of aerosols in the primary coolant system. During degraded core accidents, a portion of the fission products and structural materials will vaporize and mix in gaseous form with the coolant atmosphere. The gas phase will be a mixture of super-heated steam and hydrogen. The transport of the vaporized material from the hotter regions to nearby cooler regions will occur where condensation creates small aerosol particles. The potential escape of these aerosols from the reactor system is an important consideration in evaluating a degraded core accident. In these tests high concentration aerosols with the concentrations exceeding 100 g/m^3 are being studied. These studies are attempting to simulate real-accident scenarios in near full-scale geometry.

The basic arrangement of the facility models a PWR with aerosol transport through the pressurizer and relief tanks. Modifications of the basic arrangements will be used to model other flow paths of interest in BWRs as well as those for PWRs.

Fissium, a non-radioactive fission product simulant containing CsI, CsOH and Te, and sometimes including corium, which contains core materials subject to vaporization, are being vaporized and injected into an LWR primary system test arrangement. The tests are also being carried out with mixtures of corium and fission to study the retention of the volatile species by interactions with the less volatile aerosols.

The first three tests conducted at Marviken facility have been analyzed using RAFT. RAFT has been found to predict the particle size distributions which agree with experimental data both qualitatively and quantitatively. Very interesting data were generated when the second test was compared with the first test. In the second test the CsI feeder malfunctioned and could inject CsI at only 5% of the planned rate. The planned injection rates of CsOH and Te were achieved and were the same as in the preceeding test. Surprisingly, the measured mass-mean diameter in the second test was higher than in the first test even though the initial fissium concentration was lower. The same trend has been predicted by RAFT. The important difference between the two tests is that whereas CsI self-nucleated in the first test, CsOH does in the second. The two species have different nucleation parameters (dew point temperatures, surface tension, molecular weight). More importantly, the nucleating species (CsOH) in the second test had higher initial concentration than CsI which nucleated first in the first test.

In summary results to date show that (1) significant aerosol retention in the primary coolant system is possible, especially when water pools or condensing conditions are encountered and, (2) aerosol particles can agglomerate to large size, increasing their deposition rate even if they should be transported to the containment building.

4.4 THE LWR AEROSOL CONTAINMENT EXPERIMENTS

Even if radionuclides should escape the primary cooling system of a nuclear reactor, other barriers exist which can further attenuate them. In particular the wet steamy conditions in reactor containment and auxiliary buildings can reduce radioactive concentrations.

The LWR Aerosol Containment Experiments (LACE) are investigating aerosol processes, such as steam condensation, which are effective in removing radioactive aerosols from containment and auxiliary building atmospheres. In addition to studying the behavior of aerosols in the containment building, the program is studying the passage of aerosols through the long pipe and around the bends which would be involved in the "V" sequence accident. The LACE program is helping to clarify the currently most significant risk-contributors to off-site consequences in a severe accident. These accident sequences involve the potential for containment by-pass and early loss of containment integrity. The experiments are being performed at Hanford Engineering Development Laboratory's CSTF facility, a large $850m^3$ tank, which is full-scale for many accident scenarios of interest. This program will also involve appreciable support from other organizations besides EPRI.

The base program consists of large-scale integral tests which will simulate unique postulated accident sequences, yet which are performed to provide information on specific phenomena required to understand radioactive aerosol behavior. The tests focus on three postulated accident situations: (1) containment by-pass sequences, (2) early containment leakage or failure to isolate, and (3) delayed containment failure. A simplified test matrix is shown in Table 4-1.

TABLE 4-1

SIMPLIFIED LACE TEST MATRIX

Number of Tests	Simulated Accident Failure Mode	Phenomena Studied
5	Containment by-pass through cold leg interface piping	Aerosol retention in pipe, auxiliary building, and leak path
2	Late containment leakage due to overpressure	Aerosol containment behavior, resuspension
1	Failure to isolate containment	Aerosol containment behavior, leak path retention
1	Early containment leakage due to overpressure	Isentropic expansion effects
1	Failure to isolate containment	Aerosol behavior in intercompartment flow, leak path retention

The containment by-pass test series consist of three scoping tests and two follow-on tests. The scoping tests have been initiated prior to completion of the final LACE program plan in order to obtain early information regarding deposition of aerosols in the interface piping and auxiliary building. The tests used a 63-mm diameter pipe with soluble (NaOH) and insoluble ($Al(OH)_3$) aerosol simulants. The test conditions were chosen so that the tests could be completed quickly without additional development effort and with existing aerosol generation and analysis experience. Parameter changes considered for the follow-on tests are pipe diameter, gas velocity, and more prototypic aerosol materials. Aerosol retention in an auxiliary building and leakage paths from the auxiliary building are being studied. Appropriate thermal hydraulic conditions are being modeled.

5.0 RADIONUCLIDE REMOVAL BY POOL SCRUBBING

In many of the accident scenarios the steam-hydrogen mixtures leaving the degraded core region and passing through the primary coolant system will pass through liquid water. In assessing risk the Reactor Safety Study (WASH-1400) assumed a Decontamination Factor (DF) of 100 for unsaturated (below boiling) water pools and of 1 for steam saturated (boiling) water pools. It was believed that higher DFs were appropriate. EPRI has undertaken experimental and modeling programs to obtain the needed date(2).

5.1 POOL SCRUBBING EXPERIMENTS

The experimental program consists of three phases: Phase I involves single orifice experiments; Phase II tests use multiple orifice configurations; and Phase III involves downcomer and vent configurations.

Each phase consists of two parts: hydrodynamic tests and pool-scrubbing tests with non-radioactive stand-ins for fission product aerosols. The hydrodynamic tests have been completed(3, 4). The Phase I scrubbing tests also have been completed(5). The scrubbing experiments have been done using three aerosols: cesium iodide, tellurium, and tin oxide.

The aerosol is injected into a steam-and-noncondensible flow stream, which is then injected into the water pool in one of two tanks. Measurements are made of the amount of fission product aerosol in the injection stream, in the water pool, and in the atmosphere above the pool. The parameters investigated in the scrubbing tests include injector submergence, injection velocity (or mass flow), several aerosol particle sizes, aerosol-type (hydroscopic vs. non-hydroscopic), non-condensible gas (air, helium), steam mass fraction, pool temperature (subcooled vs. saturated), and injection geometry (single orifice, multiple orifice downcomers, horizontal side vents).

An overview of the phase 1 scrubbing results is presented in Figure 5-1. This is intended to show how the data cover the parameter ranges of interest. In this figure the range for each of key parameters that affect the scrubbing DF is subdivided into a small number of regions. Each box in the chart, therefore, represents a well-defined region of the parameter space. The crosshatched areas represent regions of the parameter space that have not yet been populated with experimental data points.

The two parameters which dominate the overall behavior of the subcooled pool are the aerosol particle size and the carrier gas steam mass fraction. The effect of the increasing steam mass fraction is most apparent for the CsI and TeO_2 aerosol material which have an AMMD in the neighborhood of 0.4 μm. The effect is less apparent for the tin powder aerosol which has a 2.7 μ AMMD apparently because the DF for this material is large even when the carrier gas contains no steam.

The effect of submergence on DF for the submergence range tested is most apparent in the tin powder low steam mass fraction data. On the other hand the effect of submergence on the DF for the 0.4 AMMD CsI and TeO_2 aerosol is less dramatic although there does appear to be a significant trend for the low steam mass fraction runs. A greater variation with submergencies would be expected if pools of

INJECTION CONFIGURATION	AEROSOL	TEMPERATURES / STEAM MASS FACTORS / FLOW RATE (g/s) / SUBMERGENCE (m)	AMBIENT									HOT POOL								
			0 02			03 08			07 099			0 02			02 05			07 099		
			0 5	5 10	10 15	0 5	5 10	10 15	0 5	5 10	10 15	0 5	5 10	10 15	0 5	5 10	10 15	0 5	5 10	10 15
SINGLE HOLE	CsI and Te	0 1 0 3	2 2 5 8 2 4 2 0 2 8*	2 1	1 4	6 3		18		200	18									
		1 0	6 0 2 7 2 2 13 2 3	8 1 4 2 4 8	2 4	8 8	13		250 60	90 890										
		1 65	5 0 11 0	2 4	2 7	8 7	140			1600 800		7.7 8.6 5.0 7.6			13			22		110PI 21
		1 65 5 8																		
	Sn Powder	0 3	110	480					300 >610 3300	>4100										
		1 0				1300	>2100		>1090	>880										
		1 65	>5600 3700			2300 >3600			>2500		3400									
		1 65 6 8																		
OTHER CONFIGURATIONS (MULTIPLE HOLE DOWNCOMER, HORIZONTAL VENT)	*Contd 3 1 2 8 2 3 1 5 2 0 5 9 2 1		PHASE II AND III SCRUBBING																	

+ Very low submergence

Figure 5-1

depth typical to BWR suppression pools are used as will be the case in the Phase 2 Pool Scrubbing Experiments. The effect of the gas mass flow rate on DF appears to be relatively weak although there is some evidence for an entrance zone impaction effect in the low submergence and low steam tin powder data.

At this point the hot pool DF data appear larger than corresponding data taken with all non-condensible gas. The most significant observation that can be made concerning these data is that the experimental DFs are higher than expected, and this observation appears to be surviving continued scrutiny by the experimenters.

The conclusions that can be drawn from analyzing the trends in the experimental data are that for the range of experimental conditions examined the pool scrubbing DF is large if the steam mass fraction of the carrier gas is large and/or the aerosol AMMD is greater than about 1 μ. It appears on the other hand, that small DFs occur only when the steam mass fraction of the carrier gas is less than about 0.7 <u>and</u> the aerosol particle size is in the 0.4 μ AMMD range. The hot pool DF's are larger than we originally expected and will certainly lead to decontamination factors for severe accidents significantly greater than "1" assumed by WASH-1400.

5.2 MODELING

EPRI has developed a computer code called SUPRA (Suppression Pool Analysis[6, 7]) which describes the scrubbing of fission products in water pools. SUPRA divides the scrubbing analysis into four zones (see Figure 5-2): the injection zone, the mid-pool zone, the pool surface, and the pool containment. The injection zone describes globule formation and breakup. The mid-pool bubble rise zone models the condensing or evaporation of steam-and-noncondensible bubbles. The pool surface zone treats desorption and evaporation from the pool surface along with entrainment of liquid droplets. The containment compartment models the removal and dilution of aerosols, and wall condensation. The model assumes that aerosols are present in trace amounts that do not affect the conservation of material and energy equations for the gas phase (water and single noncondensible gas). Also water pool conditions are assumed homogeneous. The model includes conservation equations (mass and energy) for steam and noncondensible gas mixtures, and a conservation of mass equation for the aerosol. Removal mechanisms modeled include sedimentation, inertial deposition, convective deposition, Brownian diffusion, thermophoresis, and diffusiophoresis.

In the injection zone, aerosol scrubbing from inertial and convective deposition were modeled for bubbles slugs or jets.

Calculations were performed and compared with the experimental decontamination factors and mass of aerosol scrubbed in the pool. In excess of forty cases were carried out covering a wide range of flow and pool parameters.

Figure 5-3 shows a comparison between the calculated and experimental decontamination factors (DF) for subcooled pools and pure air or helium. The calculated values of the DF's ranged from 1.33 (25% of the injected mass was scrubbed) to 3.6 (72% of the injected mass was scrubbed). The experimental values ranged from 1.4 (29% of the injected mass was scrubbed) to 13. (92% of the injected mass was scrubbed). The bulk of the data indicated a DF below 7.1 (86% of the injected mass was scrubbed). The standard deviation in the reported DF is between $\pm$ 0.4 and $\pm$ 2.7. These low decontamination factors arise because these are worst case conditions i.e., very small particles and no condensible material in the gas phase.

Figure 5-4 displays comparisons between calculated and experimentally inferred decontamination factors for steam-air mixtures injected into subcooled pools. The experimentally reported values varied between 2.5 and 2500, while the predicted DF varied between 2.1 and 5×10^4. The trends in both the data and predictions are such that the higher the steam mass fraction, the higher the DF. This is mainly due to aerosol removal by convection due to steam condensation. For steam mass fractions less than 0.15 the DF's are slightly higher than those for pure noncondensibles. On the other hand, for mass fractions of steam greater than 0.5 the DF's are markedly increased. It should be noted that the difference between the experimental value of DF = 2500 and the predicted value of DF = 5×10^4 is less than 0.04% in the mass scrubbed. For a DF of 2500, 0.996 of the mass was retained in the pool, while for a DF of 5×10^4, 0.99998 of the mass was scrubbed. The differences between the predictions and experiments are well within the accuracy of a reasonable engineering experiment or calculation.

Parametric calculations were carried out to determine the effect of key parameters on the different decontamination factors. The parameters included the particle size, steam mass fraction and pool water temperature. The calculated points connected by trend lines in Figures 5-5 and 5-6 show the effect of aerosol particle size and steam mass fraction on the DF, for subcooled (T_L = 300 K) and saturated (T_L = 370 K) pools, respectively. The results show that the DF increases markedly with an increase in the amount of steam. The minimum DF occurs around $r_p \approx 0.1$ μm (for cesium iodide aerosol). For smaller sizes, removal by diffusive mechanisms is effective, while for larger sizes removal by intertial deposition and sedimentation is effective. The calculations were carried out for conditions appropriate for a BWR suppression pool.

The effect of increasing the water temperature is to reduce the predicted DF. For a sub-cooled pool the minimum DF, at a particle radius of .1 μm, for pure air is about 13, while the corresponding hot pool value is 3.2. Evaporation into the bubbles inhibits the removal of aerosols by both convective and diffusiophoresis mechanisms. At high steam mass fractions the DF of a hot pool is large due to condensation. The effect of hydrostatic pressure causes the pool to be subcooled at the injection depth, and therefore condensation takes place.

6.0 HYDROGEN BEHAVIOR IN REACTOR CONTAINMENT

During a degraded core accident, Zircaloy will react with water to form hydrogen. This hydrogen can eventually be released to the reactor containment building. At TMI-2, it is believed that about 370 kg of hydrogen (8 volume percent) accumulated in the containment building before undergoing combustion on the afternoon of March 28, 1979. Combustion of hydrogen from the Zircaloy-water reaction presents a hazard and may affect the performance of equipment.

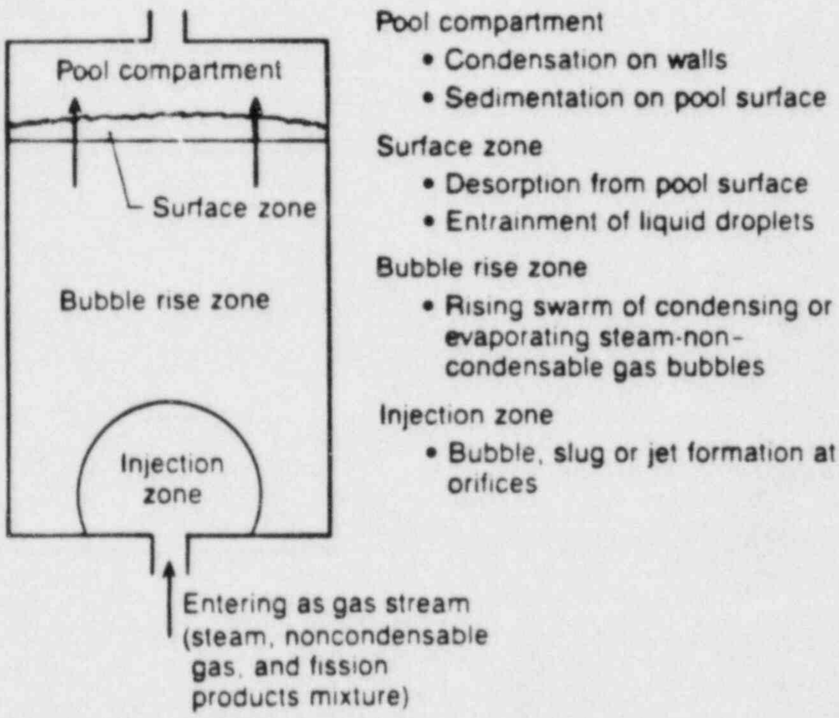


Figure 5-2. Schematic of the Pool Model

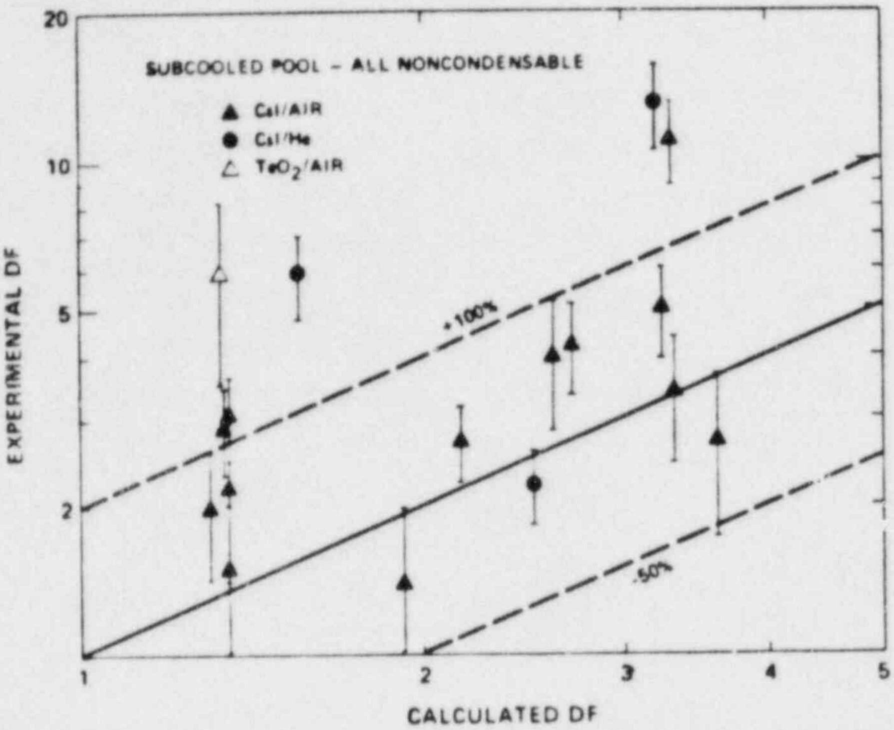


Figure 5-3. Comparison between calculated and experimentally inferred integral decontamination factors for subcooled pool and pure noncondensibles.

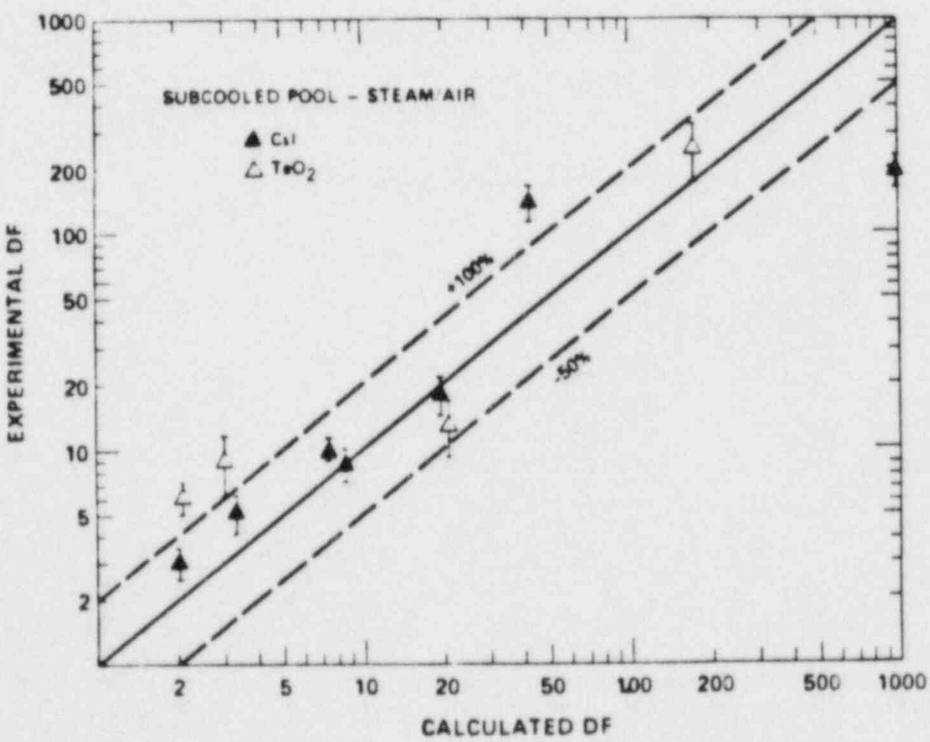


Figure 5-4. Comparison between calculated and experimentally inferred integral decontamination factors for subcooled pool and steam-air mixture.

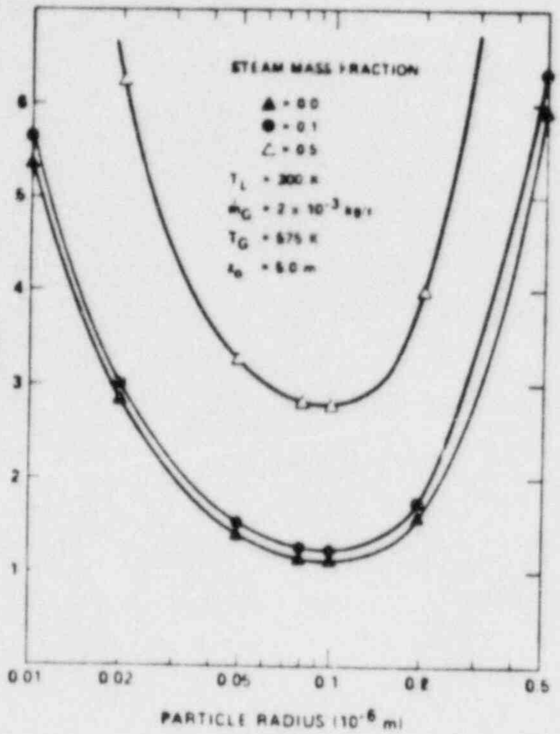
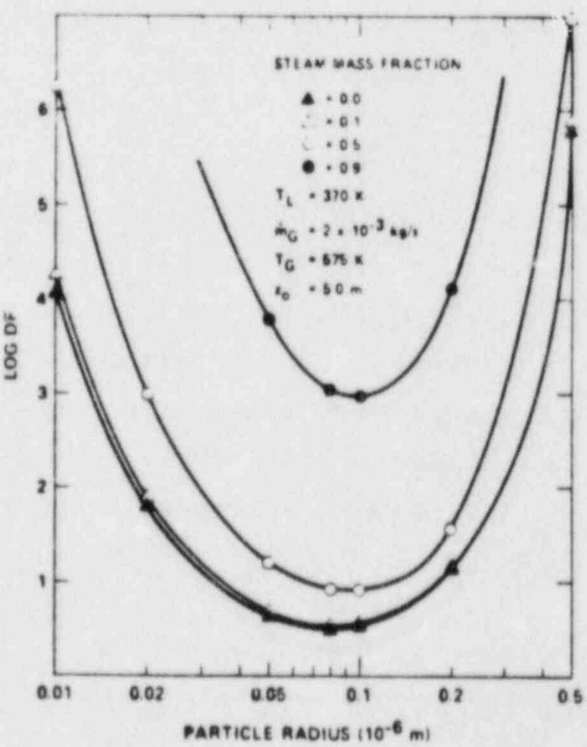


Figure 5-5. Predicted differential decontamination factors vs. particle radius as a function of steam mass fraction for subcooled pool.

Figure 5-6. Predicted differential decontamination factors vs. particle radius as a function of steam mass fraction for hot pool.



EPRI began an active program of experiments and analyses in 1981 to investigate hydrogen mixing, combustion and control for application to nuclear containment buildings (8-11). Early work was co-funded by the PWR ice condenser owners (Duke Power Company, Tennessee Valley Authority, and American Electric Power Service Corporation) and later work has been co-funded by the U.S. Nuclear Regulatory Commission and six international organizations (BMFT, Electricite de France, Kansai Electric Power Company, Ontario Hydro, Swedish State Power Board, and Taiwan Power Company). BWR Mark III owners are also participating in a Hydrogen Control Owners Group program. Completed experimental work has utilized test volumes ranging in volume from 0.6 to 74,000 ft^3, and the results have supported the development and validation of a computer code for combustion (4). Information has been obtained on the pressures, temperatures, heat flux, completeness of combustion, the effects of ignition location, hydrogen and steam concentrations, hydrogen and steam injection rates, the effects of fan-induced or spray-induced turbulence, and the potential for hydrogen stratification or pocketing.

In a test program recently completed, utilizing a 52-ft. diameter vessel at the DOE's Nevada Test Site, forty experiments were performed of two basic types. Twenty-four involved "pre-mixed" atmospheres of hydrogen, steam, and air-ignited at various locations. Sixteen consisted of "continuous injection" tests with igniters activated prior to the hydrogen/steam injection. Tests of the pre-mixed type simulate accidental ignition as in a large dry containment, and provide data for validation of models used in the analysis of degraded core accidents for both large and intermediate-size containments. Tests of the continuous type study the concept of a deliberate ignition approach to hydrogen control as employed in PWR ice condensor and BWR Mark III containments.

This work has provided a much better understanding of potential hydrogen combustion behavior. Although the phenomena are complicated, this research has shown that the threat posed by hydrogen to the containment building is less than previously thought. Flame speeds are very low, generally a few meters per second -- far lower than the sonic velocities which are required to support detonation. In the accident simulation tests, a continuous burning occurred without severe pressurizations.

7.0 CONCLUSIONS

This paper has outlined some of the experimental programs now being undertaken by EPRI. We are convinced that this information will support a lowering of the overall source term. The logical reaction to this lowering would be then to reduce the sizes of the emergency planning zones and other related requirements.

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