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MANAGEMENT OF NATIONAL NUCLEAR POWER PROGRAMS FOR ASSURED SAFETY

August 11 - 23, 1985

Conducted by

STANFORD UNIVERSITY

Under the sponsorship of

U.S. DEPARTMENT OF ENERGY
U.S. DEPARTMENT OF STATE

Presentations from the Conference

on

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Edited by

Thomas J. Connolly
Professor of Mechanical Engineering
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PREFACE

Well before Chernobyl, it was clear that there existed a strong interdependency of nations in the construction and operation of complex engineered facilities, including nuclear power plants and their ancillary facilities. It was in recognition of this fact that the United States Department of Energy and Department of State sponsored the presentation of this conference on the campus of Stanford University in August, 1985. The objective was to provide nations planning or developing a nuclear power program an opportunity to share in the American experiences, the lessons learned and the mistakes made in hundreds of reactor-years of nuclear power plant operation. The approach was to invite a number of U. S. nuclear power professionals, who were recognized for the successes they have achieved in the discharge of heavy responsibilities in nuclear power plant construction, operation and regulation, to meet with guest participants from several countries. The idea was not to say, "Here is how it should be done," but rather, "These have been our experiences and these are the procedures and techniques that have been most successful for us." The guest participants were officials who bear substantial responsibilities for the nuclear power programs of their respective countries. The opportunity was provided for the various participants to interact and to pursue questions at a very detailed level.

Although many persons and organizations contributed to the meeting, a few are deserving of special note. First is the Honorable James L. Malone, Assistant Secretary of State for Oceans and International Environmental and Scientific Affairs. It was he who originated the idea of such a conference, laid the groundwork for it and persevered in the numerous arrangements that had to be made. He was ably assisted in these matters by Dr. Charles M. Newstead, who attended to the problems of financing and orchestrated the sometimes delicate arrangements for the attendance of the participants from the several countries. Mr. James Vaughn, Acting Assistant Secretary of Energy for Nuclear Energy, addressed the closing session on the role of the Department of Energy. Brookhaven National Laboratory made funds available for the production of this record.

Part of the program involved presentations by some of the American experts. The 13 papers in this volume were taken from those presentations. Most of the papers were transcribed from an audio record. No attempt was made to convert them to grammatically eloquent papers. The flavor of the talks and the general tone of the conference are better captured in the casual attention to rules of grammar and the informal asides of the speakers. Typographical errors are, of course, the responsibility of the editor.

The papers of W. G. Counsil, L. Manning Muntzing and Chauncey Starr were simply copied from a written paper or report submitted. The paper on St. Lucie 2 is a copy of a submission of the Florida Power & Light Company to the Florida Public Service Commission, authored by J. W. Williams, Jr., a Vice President of the Company. At the conference, the presentation on construction of St. Lucie 2, covering essentially the same material, was made by Robert Uhrig, also a Vice President of FPL.

The topics of the papers divide roughly into the following groups: the organization of nuclear utilities for the management of nuclear power plant construction and operation; the regulatory approach to a safe nuclear power program in the United States; the training of nuclear facility personnel; the application of probabilistic risk assessment (PRA) to the management of nuclear plants; the kinds of assistance available to nuclear plant managers, particularly at the international level.

In addition to this written record, a number of video tapes of the presentations at the meeting have been produced. A list of these follows. Although many of these have the same topic and speaker as the papers in this document, there are some notable exceptions. These include descriptions of the construction of the River Bend Nuclear Station and of St. Lucie 2, by John Landis and Robert Uhrig, respectively; a presentation on the diagnostics for nuclear safety by Thomas H. Pigford; a discussion of the applications of probabilistic risk assessment by Ian B. Wall; a 2-hour panel discussion of seismic considerations in nuclear plant location and design, with panelists Allin Cornell, John Landis, Haresh C. Shah and Carl Stepp. At this writing, it has not been determined whether or how these video tapes will be distributed.

The many U. S. experts who attended this meeting represent a remarkable depth of experience in one or another area of nuclear power generation. While the specific audience they addressed was the group of foreign participants, much of the content of their presentations should be of interest and value to the broader nuclear energy community. The presentations reveal a commitment to doing the job right, a deep respect for the responsibilities that the profession entails, a concern for public safety and, most definitely, pride in the accomplishments. In sum, this record reveals a professionalism on the part of both regulatory and industrial representatives that is rarely perceived by outsiders, even those sympathetic to the nuclear enterprise. It is with the hope that some of this reality will be conveyed to a larger audience, both inside and outside the nuclear energy profession, that this record has been prepared.

Thomas J. Connolly
Editor
October, 1986

PRESENTATIONS ON VIDEO TAPES

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Wisconsin Electric Power

CONSTRUCTION OF THE RIVER BEND NUCLEAR STATION

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Stone & Webster Engr. Corp.

CONSTRUCTION OF THE ST. LUCIE 2 NUCLEAR POWER PLANT

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TRAINING OF NUCLEAR FACILITY PERSONNEL

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The Pennsylvania State University

THE ROLE OF THE INSTITUTE OF NUCLEAR POWER OPERATIONS (INPO)

Robert Patrick McDonald
Senior Vice President
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INPO'S PERFORMANCE-BASED TRAINING PROGRAMS

Forest J. Remick
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DIAGNOSTICS FOR NUCLEAR SAFETY

Thomas H. Pigford
Professor of Nuclear Engineering
University of California at Berkeley

THE U. S. NUCLEAR REGULATORY COMMISSION

FUNCTION AND PROCESS

Victor Stello, Jr.
Deputy Executive Director
U. S. Nuclear Regulatory Commission

REGULATORY CONSIDERATIONS OF THE RISK OF NUCLEAR POWER PLANTS
(Two Hours)

Robert M. Bernero
Director, Division of Systems Integration
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U. S. Nuclear Regulatory Commission

William J. Dircks
Executive Director for Operations
U. S. Nuclear Regulatory Commission

OVERVIEW OF THE PROCESSES OF RELIABILITY AND RISK MANAGEMENT

Edwin L. Zebroski
Chief Nuclear Scientist
Electric Power Research Institute

MANAGEMENT SIGNIFICANCE OF RISK ANALYSIS

Chauncey Starr
Vice Chairman
Electric Power Research Institute

USES OF PROBABALISTIC RISK ASSESSMENT

Ian B. Wall
Senior Program Manager
Electric Power Research Institute

THE NUCLEAR SAFETY ACTIVITIES OF THE INTERNATIONAL ATOMIC ENERGY AGENCY

Morris Rosen
Director, Nuclear Safety Division
International Atomic Energy Agency

SEISMIC CONSIDERATIONS IN NUCLEAR POWER PLANT DESIGN (Two Hours)

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MANAGEMENT OF NATIONAL NUCLEAR POWER PROGRAMS
FOR ASSURED SAFETY

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NUCLEAR UTILITY ORGANIZATION

Mr. Sol Burstein
Vice-Chairman of the Board
Wisconsin Electric Power

It is people that will make successful programs. The way we organize those people makes a good deal of difference as to whether their talents and their capabilities can be allowed to render a successful program or not. You have heard us talk about the NRC and the nuclear approval procedure in the U.S., and we have talked about the process that has made it so difficult for us in this country to consider new nuclear deployment at this time. It is not because the technology is not known to us. We probably have more nuclear power reactors operating in this country than in any other country of the world. We have had more nuclear operating years of experience. But no utility is ordering a nuclear plant in the U.S. today. The reason for that is in the process. I would refer you to two documents which were not included in the list of bibliography material but that are very important to get a picture of what the nuclear power situation is in the U.S. today. One was a document published by the Atomic Industrial Forum in July of 1984, in which a group of American utility and other people analyze what they felt were the difficulties and the problems facing the revitalization of the nuclear industry in this country and whether, indeed, the industry was required or not. That was followed by a similar but more voluminous document published in February, 1985, by the Edison Electric Institute. It is a similar treatise. Both those documents agree on two very basic premises. One is that this country, like most countries of the world, will require a vital and safe nuclear power program. Secondly, it is the process or the procedure in the U.S. that is preventing that from occurring at the present time. It is worth reading for many of you from government and other organizations in other countries so that you may understand why the U.S. is in its present posture. Perhaps you will find some lessons in our experience.

Like anything else, a nuclear organization has a function. It is designed to accomplish the objectives of the business, and, like any organization, it must be arranged so that it will anticipate the needs that have to be supplied. It has to develop the arrangement that will enable it to organize and manage and control that activity. In the case of a utility that goes into the nuclear business for the first time (a situation that all utilities with nuclear programs faced within the last 30 years or so), one has to give very serious consideration as to how that organization impacts on the existing structure. It is necessary to recognize that the nuclear program, whatever form it takes, will add to the organizational arrangement. There must be a clear and precise definition of how that is to happen. People who are involved in nuclear power will receive certain special attention. The people who are involved in the ongoing business must recognize what those attentions are and must not be put in a position where they are second or third cousins and poor relations and not perhaps receiving quite the opportunities as well as the facilities and resources needed to carry out their obligations. Any change that you make in an organization must be designed to highlight what the purpose is and must indeed signal that the utility expects certain accomplishments and certain results.

The first thing that I believe a successful nuclear utility in this country has learned (and I believe it is necessary for a utility anywhere) is that nuclear power is different. You have heard some discussion of that this morning from Dr. Kouts. It is, indeed, not simply another way to make steam, as many utilities in the U.S. believed when we first entered this business; that it was nothing more than another steam generator. As a result, it was included in the same process, if you will, with some of the same disciplines and some of the same organizations.

For a nuclear utility, nuclear power represents its biggest single investment compared to anything else. I have always maintained that if you safeguard the financial investment that you have in nuclear power facilities you will automatically take care of public health and safety. While we do say that our primary concern is public health and safety, let me tell you that the best way to achieve that is to make sure you get a financial return on your investment and that you do not risk that investment unnecessarily.

One of the things that is happening in the country, as you know, is that, thanks to the inflation of the 1970s and early '80s, nuclear power plants have gone from the 200-dollar-a-kilowatt level in the early 70s to over 2000 dollars per kilowatt at the present time. Perhaps some plants are even twice that. My company has a total capitalization of a little over two billion dollars. That is the cost of one nuclear unit some places. To what extent may I risk the total investment in my company is one facility? So we are confronted with the safeguarding of our financial resources when we say do not wager the whole company on one nuclear plant. Or if you do, make absolutely certain that there is no question about its safety and reliability.

We obviously are involved in new technology that is quite different from anything that we have dealt with in the past. Those of us in the power generation business are confronted with new demands with a less forgiving nature than we are accustomed to and certainly, therefore, with new disciplines.

In the U.S., nuclear power has taken on a visibility that is unique to it. There is no other activity that generates more emotional involvement of the public than does nuclear power. And, I must hasten to add, in this country at least, very much aided and abetted by our process and by the media. We are required by either regulation or law to make public notice of any event that occurs at a nuclear plant. If we shut down for a weekend to repair a gasket leak in a valve bonnet, we must make a public notice and it usually appears on page one of the local newspaper. Now you can have a steam plant blow up as one did in the Mojave Desert not very far from here a few weeks ago and kill six people and injure 14, and that got newspaper headlines across this country for a total of about a day and a half. We are still reading about and litigating TMI, in which nobody was hurt. The difference is visibility, the difference in communication requirements, and the difference in need as well as process demands that any nuclear organization have access to or as an integral part of itself a very competent public relations staff; I mean professional people who know how to communicate and can translate the technical jargon that many of us like to speak in among ourselves into things that the public will appreciate.

We certainly have regulation in the nuclear industry that is unlike anything else. You will hear more about that late in this program. I would make the point here that we have many different kinds of regulation, and we should not lose sight of how they each affect our nuclear business and hence how we must structure ourselves to respond to those needs. Obviously, we have the safety regulations from the Nuclear Regulatory Commission to which we have referred. Certain aspects of that also involve other agencies such as the Department of Transportation when we are dealing with waste movement by railroad or highway vehicles. In addition to the NRC, we have environmental regulations, some of which the NRC enforces on behalf of other agencies. We have the Environmental Protection Agency at the federal level in this country. In addition, each of the 50 states has its own environmental requirements, which are imposed in many cases by the local concerns dealing with such matters as thermal discharges or other non-radiological aspects. Furthermore, under our system, many of our states sign agreements with the federal govenment whereby they take over part of even the radiologically related environmental aspects of regulation at the local levels. Above all, we have economic regulation in which many states require that approval to construct the plant be first obtained before any work is done. The need for the plant, its cost, its location, and many other aspects of the plant, but particularly the economic aspects, must be approved and a license or permit issued. In many states these are called certificates of authority or certificates of public convenience and necessity. Whatever their label, they are to ensure that the costs associated with the nuclear plant will produce something that has value to the customer who will ultimately pay for the service. On that basis, these state commissions get into many aspects of the design and the construction (in theory they are excluded from all nuclear safety concerns because that is preempted by our federal government) and the economic needs and customer requirements and how those may best be met. Most recently many utility commissions have had a two-step process. They first approve the plant for construction and then, when the plant goes into service, they determine how much of the plant cost is appropriate to be included and recovered in rates that the customer is charged. Because commissions are political creatures (they have either been elected by the public or they have been appointed to their positions by an elected state official, such as the governor), they are very much sensitive to political concerns. That sensitivity has taken the form of attempts to keep rates low. I have never seen a candidate for a commission job or office advocate higher rates. Nobody has run on a platform that says "I am going to assure that you get adequate power; it's going to be the best, most reliable and safest, and I'm going to raise your rate to do it." That doesn't get you elected. So they all come out and say, "I'm going to keep your rates low." How do they do that? Well, there are several techniques. First of all, they don't allow you to charge anything for the plant until it goes into commercial service. That not only defers recovery of the cost as construction progresses, but it also defers all the interest on those charges. As a result, by the time you get done with a 2000-dollar-per-kilowatt plant, you've also got 2000 dollars per kilowatt of interest charges on the monies that have been paid out over the time of construction. If that time is extended by delays of one sort or another, then obviously the interest charges can multiply very rapidly, because these are compounded annually. So that many times utilities must go to their commissions and ask for substantial rate increases when a large, expensive project like a nuclear plant comes into service. In this country we give it the term "rate shock", when you suddenly are compelled to face the reality that you have not been paying for the previous five, ten, or fifteen years. Then rate regulators, faced with the final payment, come back and say to the utility, "Oh, but you should have done things differently. You should have built it in a shorter period of time. You didn't have to do this engineering. You should have left out that cooling water system and put in this one." All that is known as the prudency test. Was everything that you put into that nuclear facility done in the most prudent manner possible? With the perfection of hindsight, I am sure that there are a few things that even in the best of programs could have been done differently.

It is the relationship between regulator and licensee that has determined much of the manner in which the utilities have responded. The scope of regulator involvement has changed dramatically in the United States and hence has changed our needs for organizational and personnel requirements. Twenty years ago when I was involved in the licensing of some small 500-MW reactors in Wisconsin, for example, we did not have a separate licensing staff. We had one lawyer. He happened to be an in-house attorney. He did other things at the same time. We handled the entire licensing proceeding as part of the normal headquarters staff work. It did not require very much time. The hearings took one and one-half days. Today, a similar case, a similar plan in a similar location would take up a staff of lawyers, an entire staff of licensing personnel, and I'm sure it would be years. I came to San Luis Obispo, here in California, in 1966, to sit in on the first public hearing for the Diablo Canyon nuclear plant. So I could learn how to do it in Wisconsin. Well, we went back to our place and we did our thing and we built the plant, and it's been running for 15 years, and Diablo Canyon has just been licensed within the last year. Those are the differences that one can run into in terms of different environments, different time frames, and different scopes of regulation. The scope of regulation has changed dramatically in this country from one where promotion and regulation was part of the same activity. The Atomic Energy Act, as it was originally passed until the mid-seventies, had both the promotion and the regulation of nuclear power as part of its charter. So there was one arm of the government charged with both these obligations that suggested that it would be a good thing if we had this technology deployed and, secondly, deployed in a very safe way. We have, of course, changed that. We used to have one committee in the Congress that talked about all nuclear activities. It was called the Joint Committee on Atomic Energy. Now we have dozens, each of which wants to take on the responsibility of legislating new nuclear initiatives, and that makes for an entirely different prospect. We have had some units of government in this country that have been the owners of and operators of nuclear power plants; the Tennessee Valley Authority is perhaps our prime example. That is not a private investor-owned utility. It is a public entity. It is, in effect, a governmental entity. Its relations with the regulators, however, are no better, I might add, or no worse, than those of independent investor-owned utilities, at least in the United States. The Sacramento Municipal Utility District, here in California, is another example of a publicly owned governmental-unit-owned nuclear utility. There are other examples, as you know. Owning or operating a nuclear utility by government or by private activity does not really change the regulator framework nor the structural needs to accommodate that in this country.

We do have a different role of the public in the United States compared to a number of other countries. Here we have the adversarial system, in which witnesses are brought forth to support their application. They are sworn as witnesses in a trial before a jury, and they can be cross-examined. There is another form of hearing that we've been advocating and one which we hope will be adopted. That is the legislative type proceeding in which people who support the application will come in and present their cases to support the affirmative position that the utility or the advocate wants to take. Then the opponents come and they present their cases to the panel of judges or whoever is hearing the application, and they have their say. There is no cross-examination of witnesses. You can say whatever you want to. You can call that fellow an incompetent. You can say that he has no regard for public health and safety. Whatever is said goes into the record, the judges retire and they take a vote. I believe that is the system used in France, for example, where there is no adversarial approach. Everybody has his opportunity to attend the public hearing and then, as one Frenchman told me recently, "We do it the French way. We decide to build a plant and we go ahead and build it." They go through that public process, thereby not denying public participation, but perhaps giving it a different emphasis than we do in the United States. That difference imposes different demands on the organization.

The nature of the nuclear plant contract probably has as much impact on what kind of an organization is required as anything else. As you probably know, in this country some of the initial large nuclear plants were built under turnkey contracts. This was really the breakthrough in the deployment of nuclear activities in this country. In 1963, when the General Electric Company proposed to furnish the Oyster Creek nuclear plant completely engineered and constructed for a fixed price of 92 dollars per kilowatt, that was a major breakthrough. Shortly after that, there were a number of others -- Carolina Power and Light, my own company, Rochester Gas and Electric, Northern States Power. Several plants were built as turnkey plants in which Westinghouse or GE, at that time the two very largest American reactor vendors, took responsibility for the entire nuclear power plant up to the low side of the station stepup transformers. This included the turbines and the condensers and the feedwater heaters and the piping and the electric supply and the cooling water system. Everything else except the site and the transformation and transmission of the power. That meant that the utility did not have to provide on its own account much of the engineering and technical and construction disciplines necessary to design and to facilitate the construction of that type of plant. I am, of course, excluding the early plants like Shippingport, Yankee Rowe, and Dresden, which were done on an entirely different basis.

Basically, the utility needed only enough staff to make sure that the contractor was living up to his contract. We needed enough people to oversee the engineering and design, to make certain that we were getting what we were paying for. We did not, however, require a unique individual designer. We had enough construction people around to supervise the way they were pouring concrete or welding containment plates, putting piping together and a few other things. We certainly did not have enough to do the entire staff and hiring of crafts, writing procedures and specifications, and doing all the other things necessary to put the plant together.

So the nature of the contract was very important to us. The nuclear steam supply system manufacturer undertook this, of course, in order to get the volume of business that they felt was necessary to make nuclear power an economic manufacturing opportunity for them. Also, they wished to sell nuclear fuel fabrication. Westinghouse thought for awhile that it would be a good idea to sell nuclear fuel, that is, uranium itself, but they found out that it wasn't such a good idea ten years later.

In any case, it so developed that very few contractors were willing to take the risks involved in turnkey projects, particularly in the seventies. As I mentioned, inflation began to bother us, and we began to see vendors selling only nuclear steam supply systems. I'm sure you know the scope of that activity. The balance of the plant was left to someone else. The American architect-engineers, the consulting firms, the Bechtels, the Stone Websters, the United Engineers and Constructors, the firms you know quite well, undertook the responsibility of integrating the nuclear steam supply system and the balance of plant as they traditionally had for the fossil business in the past. And that's mostly the way things are being done in this country today.

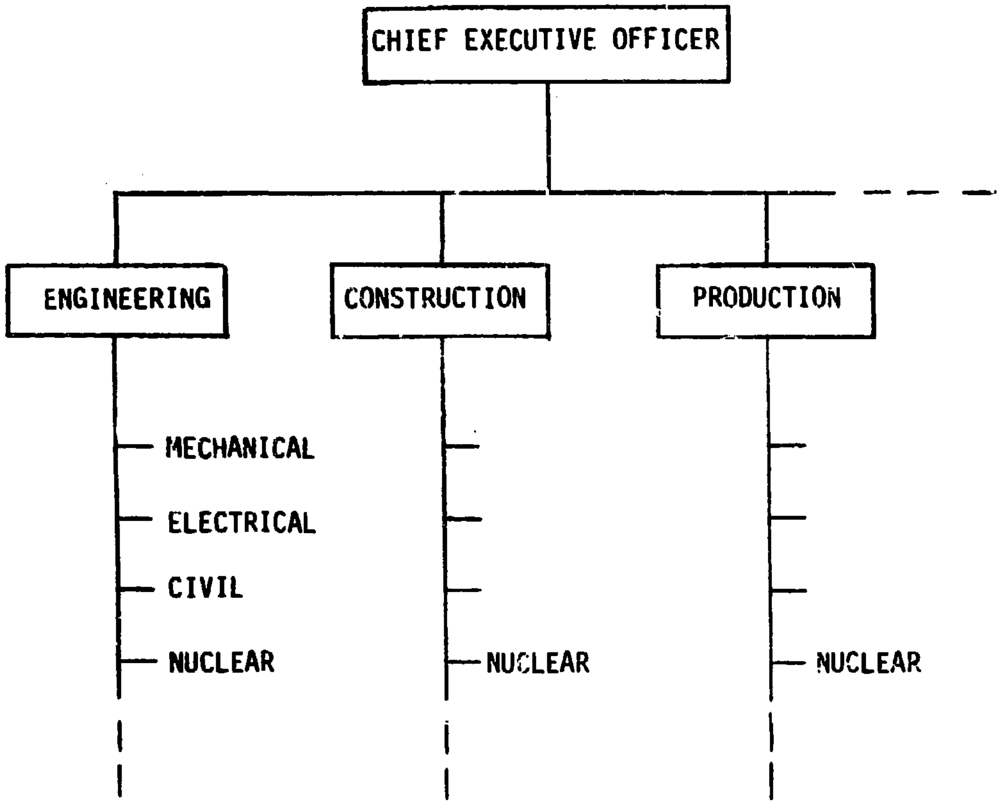
Many of us are currently engaged in the activity of trying to standardize nuclear plants, and standardization, which is a separate subject, offers many opportunities in how we dealt with the nuclear future. It also has a marked impact on the kind of organization and staff you need to do it.

Fuel supply contracts are another thing that had an impact on the organization. You could buy (and still can if you wish to) total fuel services from one supplier. They will buy all the uranium for you; they will have it converted; they will fabricate it. The only thing they won't do is take away the waste anymore. They will design the core and the reloads and get them licensed, all for the appropriate prices, thus relieving you of any staff requirements to take care of fuel analysis, fuel economics, and the fuel-reload licensing activities.

No one, however, has yet taken over the operation and maintenance of a nuclear plant. It has been suggested that if the vendors really wanted to get back in the business, they could sell the utility kilowatt-hours, and there are some contractors that are beginning to think that might be necessary to relieve the utilities from some of the risks presently attending the business.

Regardless of the contracts under which we work at present, we feel it is necessary to anticipate what will come next, because even after you have had a turnkey plant built and even after you have had nuclear fuel supply service, some day you are going to ask where you go from there with your nuclear program. Do you continue to do the same? Is this reliance the best for the utility? You will begin to plan for that time when you initiate and take more responsibility for portions of the total nuclear plant activity. This succession planning is very important.

There are probably as many different types of functional or project organizations as there are utilities. Utilities in this country are composed of prima donnas. Everybody knows best how to run the utility business. I'm no different. We all have our own experience, and hence our own preferences and prejudices. The old functional organization is the one that was traditionally used, and to some degree still is, as in some utilities in which the nuclear discipline was developed as part of everything else. You had, for example, a chairman or chief executive officer of utility and under him you had an engineering discipline and a construction discipline and perhaps a production department and some other things like fuel or procurement or other activities. Then, under engineering you had mechanical engineering, electrical engineering, civil engineering, and nuclear engineering. Somewhere down there was a nuclear department as part of the overall functional requirement of the establishment. The same was true in construction. You had somebody who built the office building, and the service center, and the services that went to each customer. You had a power plant construction department and then you had a nuclear construction group. The same was true in the production department. The first and most important thing in the production department was the system operations, your control center that controlled the flow of power over the transmission system and the amount of generation that was produced. Next were the plant production systems and somewhere under that was nuclear.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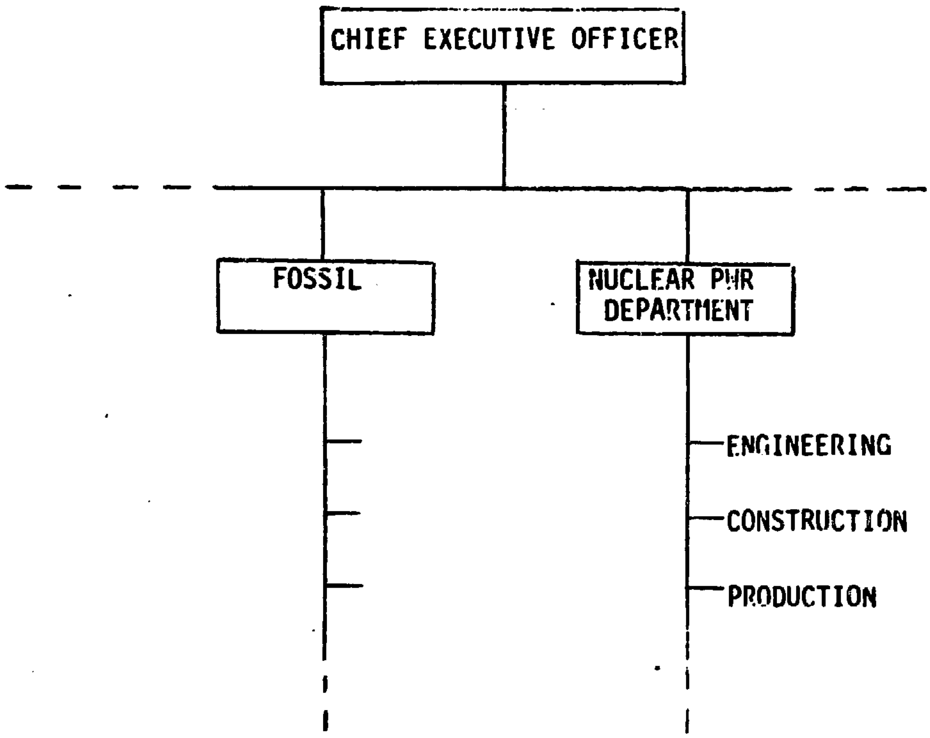
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That was the traditional system and still is in some utilities that hold to the position that they're not going to treat nuclear very much different from anything else. It's going to be a portion within the rest of the organization. That has worked in some cases. Some utilities have appointed a nuclear coordinator who is to provide some matrix functions to all of these other organizations and attempt to coordinate that activity. It's very difficult, however, for someone in the engineering department who has a vice-president of engineering who determines his salary and his hours and his working conditions, to report also to some coordinator who is supposed to tie all these things together, but who, perhaps, doesn't have very much authority. On the other hand, you have some organizations that are very large that are divided on geographic grounds. You have a company that may cover a very large area. It might have a division in the north and another in the south. Each of those divisions has its own engineering and its own construction and its own purchasing, contracting, and dealing with local regulators and so on.

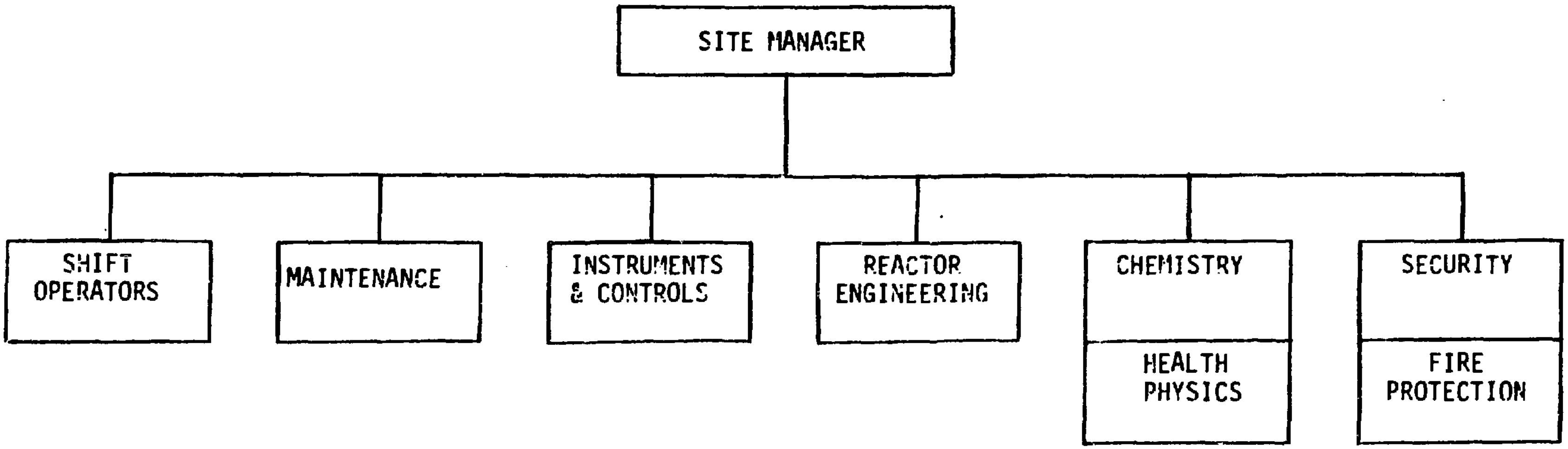
The nuclear organization that I believe has been most successful in this country is the one in which the nuclear power department is a separate activity, reporting directly to the chief executive officer. This nuclear department is usually headed by another officer under whom all of the engineering, all of the construction and operation and quality assurance for the nuclear activity are placed. These nuclear activities are located in some central place different from the fossil area and different from most of the other functions that go on. There are exceptions. It may not pay to have a separate contract department or to have a separate communication department. It's often difficult to have a separate legal department. This dedication of a separate nuclear department, reporting directly to the chief executive officer of the company, is the one that has succeeded most clearly in the United States. And it's not hard to understand why. Among other things, this promotes the most direct, short communications between the senior company management and the nuclear activity. There is a direct line; there is no interference or dilution of communication and integration of this thing with other things <u>except</u> at the top level. The president or the chairman or whoever the governing board of the utility is immediately aware of and sensitive to everything that happens in the nuclear department. Whether it be in engineering or construction, whether it be in operation and maintenance, in quality control or insurance problems, or in any other part of the activity related to nuclear. It presents a direct, interrelated communication among all the people who are involved in the nuclear activity. They can talk to each other in a relatively small staff group without having to go up through one officer and down through another one, and there is very little sensitizing of the information as it goes back and forth. It provides a focus for all nuclear activities from the outside as well as the inside of the organization. If you're dealing with the public, with the state regulatory people, with the Nuclear Regulatory Commission in this country, you have one central focal point that isn't, again, diluted or doesn't have other problems or priorities that a different organization may have. Above all, it establishes the philosophy that nuclear power is important, as important as anything else in the structure -- as important as personnel, or finance, or accounting, or line construction, or transmission and distribution. Nuclear power ranks equally with all the other primary functions that the company has to discharge. And it presents that philosophy for the world and the people inside the organization to understand and appreciate.

I guess, when we talk about the overall place where nuclear power fits into the organization, I've already started to tell you a little bit about where the power plant organization fits into the nuclear activity itself. I think it would be appropriate to talk about where that is. In my opinion, the people at the nuclear plant site have to have one person in charge, just as we see in the organization below, where we had a nuclear power department which consisted of the officer in charge, an engineering group at the headquarters, a quality assurance or quality control group that took care of the special unique features of the nuclear activities, and the operating and maintenance group. By the operating and maintenance group I refer to the power plant itself, because there's really very little activity that the headquarters staff should be doing in regard to operating and maintenance. If they do, they're probably duplicating what's going on at the power plant.

It is in this area, the organization of the plant, that two things I would call to your attention are basically important from my perspective. One is that there must be a single plant manager responsible for everything that goes on at that site. That includes some of the things that he may have little control over, but ultimately he must have the direct on-site responsibility, even including security. Often, things like security or quality assurance or safety reviews might be the responsibility of someone else but not if they occur on-site. So I would recommend that you have a site manager at every location. You can call him the general manager. You can call him the site manager. You can call him the plant superintendent, or the plant manager. Whatever title you wish to give him, make sure that he is responsible for everything that goes on in that site and that you do not dilute that responsibility. He must have, obviously, a group of shift operators, and, depending upon the nature of the plant, its control room, and functional arrangements. That can vary anywhere between four or five people per shift to 15-20 per shift.

I will say something now that I will keep repeating again and again. The number of people at the site, this is Principle #2, must be as absolutely small as you can make it. I know some plants where there are a thousand people on the site. They may be divided up into three shifts. There are 200 in the second shift, 200 in the third shift, and 600 or 700 on the first shift. Do you think he can manage that? You are asking the man in charge to do an impossible job. Or else you have to put so many layers of supervision between that man in charge and the people actually doing the work that he never knows what in blazes is going on. And he doesn't have a chance to do anything about it. The shift supervisors should be as close to the site manager as you can make it.

We used to think that we could get by with four shifts. This would give us enough to have three rotating shifts and to take care of seven days a week and holidays and sick leave. And then we went to five


SITE MANAGER
SHIFT OPERATORS
MAINTENANCE
INSTRUMENTS & CONTROLS
REACTOR ENGINEERING
CHEMISTRY
HEALTH PHYSICS
SECURITY
FIRE PROTECTION

shifts, and that took care of things. And now we have six shifts. And that sixth shift is very valuable, because it allows one shift to be in training and to upgrade their performance, or to maintain their level of knowledge and information all the time. It's often a good idea to consider five or six rotating shifts to take care of the shift operations.

Now, you'll need a few other things besides people to run the plant. You'll need some maintenance people, and these are the people who actually do the hands-on maintenance work, the repairs, some of the tests, performance tests, coupled together with the shift-operating people and coupled together with another very important thing that I think bears separation, and that's the instruments and control group. Often, in the old types of plants where we had pneumatic and hydraulic devices primarily as our instrumentation and control, we could put these in with the maintenance and the operating groups. This now requires a much more sophisticated caliber of individual. We are now perhaps as computerized in our nuclear plant operations as anything else, and I believe that this area requires some very special activity. You need some reactor engineering, and this is of course, as the name implies, to direct the technical operation of the plant from the physics point of view. Since we're dealing with the kind of machines that Dr. Kouts described earlier, this takes on some very significant means. Further, we have found that chemistry in the plant is very important. We cannot ignore some of the traditional ways we used to treat feed water, for example, and the way we handle wastes. So there are many aspects of our plant chemistry that require a different level of professional activity than previously. Since chemistry and health physics are often so closely related, it's not unusual to have a health physicist combined with the chemistry activities in a similar department.

Now, you obviously need some other functions in the plant. You probably have, as I mentioned, security and fire protection, which often go together, and you have an office, because you have to keep track of people coming and going and their payroll and the hours they work and when they don't call in, and a few other functions. So sometimes it pays to interpose an individual who will take charge of these two, and perhaps another individual who might take charge of these more technical disciplines, and have them report to the superintendent. But you shouldn't have any more than two layers between the site manager or the superintendent and the people heading up these operating activities at the site.

We can talk about some of the other basic functions that have to go on at every power plant. Training is becoming an abusive term in the U.S. Ever since Three Mile Island, as you can imagine, people have said, "The trouble with Three Mile Island is that the operators were not very well trained." The people who said this were, first of all, the other owners of B&W power plants. They couldn't very well admit that it was the design of their plant, their typical plant, that was basically at fault, because otherwise they would all be shut down. So it was easy for all the other B&W owners to say that it was the operators. This was supported by the Nuclear Regulatory Commission and all the people they hired. After all, they couldn't admit that one of their plants whose safety they had approved was inherently perhaps too sensitive to some things and should have been perhaps redesigned or received some other safety category. In any event, it was easy to blame the operators, because they had nobody speaking for them except the utility, and they were already in hot water anyway, so a little more wouldn't hurt. We're still arguing about that today. When was the accident? 1979? What is this, 1985? So you can count like me. And do you know what the only holdup to relicensing Three Mile Island Unit 1, not the one that was damaged--that will never run, in my opinion -- but Unit 1? Operator qualification. That is what courts are being asked to consider for I think the hundredth time. So training has become a very integral part, and so has quality assurance, as a result of that same related activity.

We have to consider one other thing in a plant organization, and that has to do with how you handle emergencies. I'm assuming that you understood when I said, "Keep this thing as small in personnel as possible," that during an outage, either for refueling or maintenance or something else, you may have to bring in people from some other organization to augment or to help your own staff. And that's what we do. We bring our own people from other parts of our company, or we hire contractors to come in for the month or two that it might be necessary to do this work. At Point Beach, I had to replace two steam generators after 15 years of service, and I'd be glad to discuss why when we have a moment, if you wish. We decided to hire some of the staff to help us do that job, from the outside. And we were successful. We did that job in six months and for far less than some of the other projected costs of doing the comparable work, by organizing and managing the activity ourselves, but augmenting it with the staff that we had before. Our plant was built perhaps with as high a quality as anything else, and the reason for it was I hired the majority -- all the supervisors -- and the majority of the other staff the first year we went into the nuclear business. We decided in 1965 we were going to go into the nuclear business, and in 1966 I had the site manager designated, hired, and assigned to that job, and he and the others came along immediately thereafter so that they could work with the designers during the design and could understand why these particular designs were occurring. They would need some off-site training. We knew that. So we sent them away to places like Saxon in Pennsylvania, or other plants, research reactors, wherever there was one, to try to get some experience with the nuclear disciplines that they might not have had. Many of our people came from our American Naval program. So we had some people with that kind of background. But we gave them a year or two, or whatever it took off-site, so that they could be prepared to meet their responsibilities. And then when the plant started to be built, we transferred that group to the site, and they developed their procedures and their organizational and staffing and maintenance and other requirements half the time, and the other half the time they made the <u>best</u> inspectors you ever could find. We sent them out to inspect the construction, and since they were going to live with whatever the contractors and the welders and the fitters were doing, they made sure that they obtained first-class work. The results have spoken for themselves in our case, we think, very well.

I believe very strongly that you can make any organization work if you have the right people in it. But, as I said before, it works much better if you put the people in a place where their talents can show. In this respect I think we have a very significant personnel-selection process. We select people on the basis of their capability. Some people call it I.Q., intelligence quotient. We don't want people who can't think or who have no abilities to respond to a situation; so, among other things, we also want to make sure that they have aptitudes for the job,. In addition to intelligence, we are looking to see that people will have mechanical or electrical or other kinds of aptitudes, since it will make it much simpler for them to recognize and appreciate the significance. I know many economists with 140 I.Q.s that I wouldn't let within 40 miles of a nuclear plant. That doesn't mean that they're not clever, that they're not bright, but they have no aptitude for the job that's involved. And certainly I want somebody who will have emotional stability, who will be able to take the kind of rigors and call it crises that are imposed in the nuclear operations.

I mention, before I digressed, that in addition to this organization you need to set up an organization that would handle local emergencies if you should have an event at one of these plants. People must know in advance who is responsible, who is to do something, where their responsibility starts and stops, and where they come from. So in this respect it's good to have a separate structure outlined ahead of time that will say, "If we have an incident, this is the way we will respond, and these are the people who will respond, these are the people we will notify, this is the way we will handle that particular ..." -- not prescribing the details of the response, but simply the structure of doing so. I think, as I said before, that people need to be encouraged and motivated to stay in this crazy business, and I think that one has to have some policies about personnel that go to motivating and rewarding people who are in it. Obviously, they have to be seen as being competitive with the other things that are going on within the institution, and you can't make wholesale changes; but, if you're going to entrust a $2 billion plant to somebody that you pay $5000/yr, be prepared for the consequences.

I think it is essential that you plan for the future. How long can you keep a shift superintendent on shift? Four years? In some countries the culture will permit that. In the United States we found that it won't. Succession planning in all of these areas is a very important part of providing the continuity of organization and the spirit of excitement and motivation that go with it. I hired the first superintendent at the Point Beach nuclear plant in 1965. He happened to have come from another nuclear plant (there weren't very many running in those days, you know), and we were friends, so he came because he wanted to. He has just retired. He's reached that age of mid-60s, and it's his opportunity to do something else. I have had to plan for his replacement. I could have waited until he chose to and then perhaps made a choice about who was there, but it was far better for the choice to have been made years earlier for the training to be given to that person through various opportunities of discipline that we had, and to permit him and the older site manager to transfer from the old guard to the new in a way that was smooth and easy -- not only for the individuals that were involved there but for everybody else.

The age of plants and the age of people, of course, becomes something one has to deal with as time goes on. I'm not sure that may interest many of you at this moment, but let me tell you to start planning for it when you first start planning for your plant. I don't mean to get into design areas or construction areas, but I'd like to tell you about how this impacts organizations. If you don't plan for it, you end up with much greater organizational needs than you would otherwise have. I mentioned to you that we had to replace some steam generators at the Point Beach nuclear plant. I had a feeling in 1966-67 that that was probably going to be the first big piece we would have to replace. Therefore, I made the equipment hatch big enough to get them in and out. That was then a simple job. It was a rigging job. It did not require a structural redesign of the system in order to accomplish it, and it saved us six months in the operation compared to another plant that has just replaced steam generators, where they had to chop concrete and cut lining and remove shielding, and it was a very expensive and a very

II BEFORE THE FLORIDA PUBLIC SERVICE COMMISSION
IN RE: ST. LUCIE UNIT NO. 2 COST RECOVERY

J. W. Williams, Jr.
Vice President of Nuclear Energy
Florida Power & Light Company

BEFORE THE FLORIDA PUBLIC SERVICE COMMISSION

FLORIDA POWER & LIGHT COMPANY

SUPPLEMENTAL TESTIMONY OF J. W. WILLIAMS, JR.

In Re: St. Lucie Unit No. 2 Cost Recovery

DOCKET NO. 820097-EU

TESTIMONY

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BEFORE THE FLORIDA PUBLIC SERVICE COMMISSION

FLORIDA POWER & LIGHT COMPANY

SUPPLEMENTAL TESTIMONY OF J. W. WILLIAMS, JR.

(In Re: St. Lucie Unit No. 2 Cost Recovery)

DOCKET NO. 820097-EU

<u>Professional Background</u>

My name is J. W. Williams, Jr. My business address is 9250 West Flagler Street, Miami, Florida.

I am a graduate of the University of Florida with a degree in Chemical Engineering. I have completed the Executive Program from the Graduate School of Business of Stanford University and the Advanced Nuclear Technology Program at the University of Florida. I am a Registered Professional Engineer in the State of Florida. I have been employed by Florida Power & Light Company ("FPL" or the "Company") since 1950 when I joined the Company as a Student Engineer. I have held several operating positions in power plants including Plant Superintendent for Turkey Points Units No. 1 and 2 (fossil) and Units No. 3 and 4 (nuclear).

In 1972 I was responsible for supervising FPL's compliance with Federal Quality Assurance requirements as Manager of Quality Assurance. In 1973 I became the Project General

Manager for the St. Lucie Project, and in 1975 became the Director of Projects responsible for all power plant construction. I became Director of Nuclear Energy in 1981 and was elected Vice President in 1982.

As Vice President of Nuclear Energy, I have responsibility for the operation and maintenance of all operating nuclear power plants as well as technical services for those plants.

Introduction

My testimony has three main focuses: (1) a description of FPL's St. Lucie No. 2 nuclear unit; (2) a description of the construction of that unit, including a discussion of the regulatory process and its impacts on the construction schedule, as well as a review of the FPL construction management programs aimed at controlling costs, expediting construction progress and maintaining high levels of quality; and (3) an analysis of the cost of the unit and the impact of various external factors on the originally budgeted cost estimate for the unit. Mr. Gardner has testified regarding FPL's need for St. Lucie Unit No. 2. My testimony shows that Florida Power & Light Company has acted prudently in building the unit and has completed the unit at a reasonable cost. In addition, I will address the operation, maintenance and fuel expenses which will be incurred by FPL in operating St. Lucie Unit No. 2, as well as the 13 month average of FPL's investment in the unit and in nuclear fuel for the unit, during the test period being used in this proceeding (July 1, 1983 - June 30, 1984).

Using the unit's cost and operating expense which I present in my testimony, Mr. J. L. Howard presents testimony showing the total incremental revenue requirements for the Company that will result from the first full twelve months of commercial operation of St. Lucie Unit No. 2. Mr. L. L. Williams discusses the proper basis for allocating and recovering these revenues through base rates.

Description of St. Lucie Unit No. 2

St. Lucie Unit No. 2 is a nuclear-powered electric generating facility having a net summer electric output rating of 802 megawatts. This unit is jointly situated with St. Lucie Unit No. 1 on a 1,132 acre site on Hutchinson Island in St. Lucie County, about halfway between the cities of Ft. Pierce and Stuart on the east coast of Florida.

The two main operating components of this unit are a Combustion Engineering nuclear steam supply system utilizing a pressurized water reactor, and a Westinghouse turbine generator. These components were selected as being duplicates of those used at St. Lucie Unit No. 1.

The initial budget commitment to construct St. Lucie Unit No. 2 was made in 1972. At that time, the preliminary estimate indicated the unit would cost approximately $360 million, based on the projected costs for Unit No. 1, which was under construction at the time. This estimate was based on the assumption that Unit No. 2 would be a duplicate of St. Lucie Unit No. 1. Document No. 1, however, shows that the unit was not constructed as a duplicate unit, primarily because of various regulatory actions. The original cost estimate was also based on an estimated in-service date of 1979; regulatory and other delays prior to the start of construction have shifted the actual in-service date by four years. Finally, the cost of St. Lucie Unit No. 1 turned out to be substantially higher than projected, increasing from its estimated of $318 million in 1972 to a final actual cost of ~~$400~~ 475 million in 1976.

We are currently projecting fuel loading to begin at Unit No. 2 in March 1983. The most recently budgeted cost for the unit is $1,420 million; this budget amount was approved in October, 1982. This figure does not include $ 46 million in backfit items we anticipate having to add in order to satisfy new licensing requirements. As will be explained in greater detail below and in the accompanying documents, the rising cost of constructing this unit, while quite reasonable in relation to the experience of other utilities, has been due to a variety of factors, most of which were outside of FPL's control.

The Construction Licensing Process

Essential to an appreciation of the difficulties and the potential for delays and cost increases in constructing a nuclear power plant is an understanding of the regulatory constraints under which that construction takes place. The most significant of those regulatory constraints is the construction licensing process, which I will briefly describe.

The construction licensing process for a nuclear project is both complex and comprehensive, and has become more so in the years since construction of St. Lucie Unit No. 2 was initially approved. Essentially, the construction licensing process comprises two main elements: site certification by the State of Florida, and the issuance of a construction permit by what is now the Nuclear Regulatory Commission ("NRC"). I would note that, at the time the Company initially applied for its construction permit from the NRC, there was no state site certification requirement. However, the Florida Electrical Power Plant Siting Act became effective on July 1, 1973, forcing FPL to pursue a second and independent path in the construction licensing process.

Document No. 2 details the chronology of the major events in the construction licensing process. As may be seen on this document, application was made with the NRC for a construction permit on May 14, 1973, and the site certification application was filed with the Florida Department of Pollution Control on January 23, 1974. Although the Florida Electric Power Plant Siting Act provided for the certification process to be completed in 14 months, the site was not certified until May 18, 1976, twice what the Act allowed. The NRC construction permit was not issued until May 2, 1977. FPL made some progress on the project under an NRC Limited Work Authorization ("LWA") from June-November, 1976, without a construction permit (a considerable risk to FPL, since there always existed the possibility that the NRC would not issue the permit and thus render FPL's efforts and expense under the LWA useless). However, except for this limited, short-term opportunity to proceed, FPL could do nothing toward constructing Unit No. 2 and making it available for service until the issuance of the construction permit -- four years after the application for that permit had been filed. A review of Document No. 2 will show the large number of stops and starts and the tortuous detours which licensing a nuclear facility entails. An example of this is the stop work order issued by the U. S. Court of Appeals due to an alledged deficiency in the NRC review process.

In addition to creating delays, the construction licensing process was also responsible for greatly (and expensively) expanding the scope of the St. Lucie Unit No. 2 project. The specific cost increases attributable to the expansion of regulatory requirements after the licensing process for Unit No. 2 was commenced are shown in Document No. 8, discussed later in my testimony; these increases were substantial. An idea of the magnitude of the increased number of regulatory requirements FPL had to meet in constructing St. Lucie Unit

No. 2 is provided by Document No. 3. This document shows that since 1972, there have been over one thousand new formal NRC regulatory requirements issued which have had an impact on the construction of nuclear power plants like St. Lucie Unit No. 2.

The new and revised regulatory requirements have not only resulted in specific scope expansion for the unit, but have also added greatly to the manpower and costs needed to complete all the work within the original scope of the project as well. These "indirect costs" are largely the result of the rigid quality control and inspection requirements which the NRC has imposed, together with the massive documentation necessary to demonstrate that the requirements have been satisfied. All of these factors distinguish construction of a nuclear plant from construction of less-regulated coal or other fossil fueled units.

Of course, FPL has not passively accepted these regulatory delays and expanded facility requirements. On the contrary, the Company has aggressively and diligently pursued all available means of cutting through the red tape and hastening the licensing process. A good example of this is the handling of the Final Safety Analysis Report which FPL submitted for review as a part of the NRC licensing requirements. We were told by the NRC that, due to manpower constraints in the NRC, we would be placed in line for our Final Safety Analysis Report review much later in time than we could accept and still remain on our schedule. This matter received full management attention and, through our actions, we were able to cut the review time for the Final Safety Analysis Report from the normal eighteen months to just seven. Moreover, we did this without affecting the thoroughness of the review. Nevertheless, FPL's considerable efforts could only reduce, certainly not eliminate, the impact of the ever-expanding regulatory requirements on the cost and schedule for construction of St. Lucie Unit No. 2.

<u>The Construction Process</u>

In contrast to letting the contractor manage the construction at the project with little involvement by the utility, FPL undertook the lead role in construction management at St. Lucie Unit No. 2 from the very outset. This has allowed us to remain in constant control of the project, and to make and implement quickly the management decisions needed to respond to changing regulatory requirements. We feel that our project management at St. Lucie Unit No. 2 has been both innovative and effective; as evidence that others view it similarly, we have been, and still are, asked by other utilities to provide information which will help them to develop similar management systems for their own projects.

Florida Power & Light follows three major principles in managing its major construction projects. The first principle is that more attention to managing and controlling the project will lead to shorter actual construction schedules and a greater likelihood that the overall cost will be lower. The second principle is that a sound formalized management and control system should be used by FPL and its contractors. The system must be flexible to accommodate changes in the project and must be used by project personnel in their control of the job. The third principle is that high standards for the quality of the design and construction of a project should be maintained throughout the duration of the project. This principle may be summarized by the phrase "do it right the first time." When changes are necessary late in the project, great disruption of the schedule and large increases in cost can result.

FPL has applied these three principles in the construction of St. Lucie Unit No. 2 to the greatest extent possible, recognizing that with a project of this size and complexity it is impossible to adhere to all project management principles at all times. I believe that adherence to these principles is largely responsible for the success FPL has had in controlling the unit's construction schedule and costs.

In order to manage effectively the St. Lucie Unit No. 2 project in conformity with these principles, FPL established a Project Team, under the control of a Project General Manager. This project management approach is explained in detail in pages 1-3 of Document No. 4, so I will describe it only briefly here. The Project General Manager has the responsibility and authority for the total management of the project from its inception. Figure 1 of Document No. 4 is an organizational chart which depicts the lines of responsibility both above and below the Project General Manager. One of the key subordinates to the Project General Manager is the Site Manager. The organization under him, shown on Figure 2 of Document No. 4, consists of both utility and contractor personnel. By placing FPL personnel in the key on-site support functions, we have been able to maintain much more direct control over and involvement in the construction of St. Lucie Unit No. 2 than would otherwise be the case.

There was an early emphasis placed on developing a control system for use by the project management in keeping track of the schedule and cost of the project. A variety of control tools were used to keep the project on track. These are described in Figure 3 of Document No. 4. They include various systems to measure and report on schedule progress, cost levels, productivity, quantities of materials used, materials availability status, etc. In addition to the baseline management controls, many other initiatives were taken by the Company to help control costs and maintain the schedule. These are summarized in Document No. 5. Document No. 6 gives a summary of various techniques that FPL used in the construction of the plant to incorporate the lessons learned in the industry and on St. Lucie Unit No. 1. Document No. 7 summarizes the chronology of major construction activities on the project.

Another major control on costs at St. Lucie Unit No. 2 is the Company's capital expenditures budgeting process. This process has been explained in some detail in both this and our prior rate proceedings; perhaps the most thorough description was in the prepared testimony of FPL's witness Mr. B. L. Dady in Docket 810002-EU. I would direct to that testimony anyone who requires a complete discussion of FPL's budgeting process. Briefly the process operates as follows. Major projects such as St. Lucie Unit No. 2 receive scrutiny from not only the Company's Budget Committee, but from the Board of Directors as well. Authorization for a project extends only up to specified dollar limits; if costs increase beyond the budgeted amount, approval must be separately sought for increases in the project budget. I believe this process provides a powerful incentive to those directly responsible for managing a project to see that costs are kept within budgetary limits.

While the documents I just referenced provide more detailed discussion of the various controls, initiatives and other management tools we have used at St. Lucie Unit No. 2 to help minimize construction time and costs, I would like to mention briefly a couple of specific examples of these tools in action in order to give a feel for the type of efforts FPL has made. The first example is FPL's use of time lapse photography to identify potential work methods improvements. The installation of condenser tubing at St. Lucie Unit No. 2 was analyzed using this technique and it was discovered that the tubing crew normally used for the industry for this operation was larger than necessary; only two men could effectively work on each side of the condenser at any one time. The crew size was subsequently reduced at a considerable savings in labor costs. Secondly, I would point out FPL's use of an automated weld control program at the unit. At St. Lucie Unit No. 2, there are 26 different qualifications which must be received by welders for different types of welds and over 200 welders; these welders will have made over 44,000 welds when the unit is completed. It is apparent that welding operations of this magnitude and complexity make systematic control of welding activities extremely important. Under our automated weld control program, a "traveller," or weld documentation sheet, is automatically printed for each certified weld and shows the designated hold points and special instructions for the weld as well as the initials of the qualified welder who performs the weld. As a result of this program, the savings in both construction time and labor costs that are derived from the reduction in missing hold points and/or unqualified welders have been substantial. There are many other examples of how FPL's project management system has saved time and money, but I believe these two are representative of the types of benefits which are derived from applying sound management techniques to the construction of St. Lucie Unit No. 2.

The Cost and Schedule of St. Lucie Unit No. 2

FPL believes that the current $1,420 million budget for St. Lucie Unit No. 2 is a reasonable cost. A question has been raised about the $360 million original estimate. As I stated previously, this estimate was based on several major assumptions which have proven incorrect. In particular, the duplication of St. Lucie Unit No. 1 that was assumed, in fact did not occur. Many of the economic assumptions concerning inflation and the cost of capital have also changed. I should also stress that this initial cost estimate was extremely rough and was recognized as such by FPL when made. Nonetheless, the figure of $360 million was the first budget amount approved by FPL's Board of Directors for St. Lucie Unit No. 2, and I believe it will be useful to review how the cost of the unit increased from this original estimate to the current budget amount.

As I mentioned earlier, the current budget (October, 1982) for St. Lucie Unit No. 2 excluding backfit items, is $1,420 million. My Document No. 8 shows a summary of each change that contributed to the increase from the original estimate to this figure. Each of these changes has been assigned to one or more of four categories which describe the cause or causes of the change. This type of allocation, by its nature, is judgmental; however, it provides some insight into the magnitude of the cost increases related to the various categories of causes. These four categories, and the percentage of the total cost increase attributable to them, are: (1) Regulatory Requirements (55 percent) -- changes in scope necessitated by changes in applicable Federal, State or local regulatory requirements; (2) FPL-Initiated Scope Change (10 percent) -- changes in scope related to safety, efficiency, or reliability which FPL determined were necessary in light of our experience with other units or the experience of the industry as a whole; (3) Unanticipated Escalation Changes (12 percent) -- deviations from earlier unit cost estimates for material and labor; and (4) Design/Estimate Refinement (23 percent) -- cost estimate changes due to refinements in productivity estimates, working conditions or construction methods specifications, or plant design as a consequence of increasing experience in constructing the unit.

It is apparent that by far the largest single factor in the cost increase experienced by FPL in constructing St. Lucie Unit No. 2 is the regulatory environment in which the unit was constructed; more than half of the total increases is attributable to this factor. Of course, this should not be surprising, given the lengthy delays, massive scope changes and other impacts which regulations have had on St. Lucie Unit No. 2, and which I discussed earlier in my testimony.

Moreover, I feel that it is also important to point out what the percentages shown above demonstrate about the extent to which the cost increase at St. Lucie Unit No. 2 was due to factors beyond FPL's control. Two of the categories shown above (Regulatory Requirements and Unanticipated Escalation Changes) involve external influences that are almost entirely outside FPL's control; together, they account for 67 percent of the total increase. A third category (Design/Estimate Refinement) is largely dependent on the direct impact of the scope changes reported in the Regulatory Requirements and FPL-Initiated Scope Changes categories, and on increasing design and construction experience as the project progressed. This category, which accounts for 23 percent of the total increase, ~~the final category,~~ contains many gray areas that are really second order regulatory effects that cannot be specifically identified to a particular change in requirements but nevertheless increased the complexity of the project and complicate the schedule being implemented. The FPL-Initiated Scope Changes category, comprises only 10 percent of the total increase. Most of the scope changes and associated cost increases relate to improvements which were made in order to insure that the unit would be able to operate as safely, reliably and economically as originally envisioned; it has not been FPL's intent to expand the unit beyond its original design performance characteristics. A good example of this type of scope change is the revision which FPL has made to the unit's steam generators and other secondary side components in order to incorporate the Company's and other utilities' experience in fighting corrosion of the sort which affected the original steam generators at Turkey Point.

The project management process used by FPL in constructing St. Lucie Unit No. 2 paid off in the form of truly superior performance in limiting the timetable for constructing the unit. Document No. 9 shows that St. Lucie Unit No. 2 was constructed within the same elapsed time as Unit No. 1, notwithstanding the extraordinary increase in regulatory requirements which occurred between the construction of the twc units, as shown on Document No. 3. Moreover, Document No. 10 shows that the St. Lucie Unit No. 2 construction schedule is substantially shorter than the schedules of virtually every other nuclear plant which has gone or will go into service during the first half of the 1980's (only one other unit's schedule is within 15 months of Unit No. 2's), and is 43 months shorter than the average. If St. Lucie Unit No. 2 had taken as long to construct as the industry average, the plant would have cost approximately $2 billion even if no further regulatory requirements were enacted during the additional construction period.

In addition to keeping the unit's construction on a tight timetable, we were also successful relative to the rest of the nuclear industry in controlling the unit's cost. My review of the cost of St. Lucie Unit No. 2 shows that it compares favorably to the cost of other nuclear units which have recently or will soon go into service.

In summary, I believe that we have clearly demonstrated the efficiency of FPL's project management in constructing St. Lucie Unit No. 2. This unit, which has been added at a reasonable cost, represents a valuable addition to FPL's generating resources. Although the cost of the unit is significantly higher than could have been originally anticipated, the majority of these costs are attributable to factors such as the regulatory and economic environment which are beyond FPL's control.

Plant In-Service, Nuclear Fuel and Operating Costs for St. Lucie Unit No. 2 for the Period July 1983-June 1984

As I described at the beginning of my testimony, Mr. J. L. Howard will be developing the revenue requirements associated with St. Lucie Unit No. 2 using investment and operating cost figures supplied by me. These figures are shown on my Document No. 11.

The projected plant in service dollar amount shown in Document No. 11 includes backfit items that are required by the NRC to be installed after the unit goes into commercial operation. These are included to the extent that this backfit work will be transferred to plant in-service during the time period under consideration. AFUDC has been included. The operating and maintenance figures were extracted from the Company's current 1983 operating budget and the 1984 forecast.

The nuclear fuel balances in Document No. 11 were based on contractural commitments for the fuel and on projected fuel consumption by the unit. All of the amounts reflected in Document No. 11 have been adjusted to reflect FPL's ownership in St. Lucie Unit No. 2.

TESTIMONY

INDEX OF DOCUMENTS

1. St. Lucie Unit No. 1 vs. Unit No. 2 - Scope Comparison
2. Licensing Chronology Summary
3. Regulatory Impact on St. Lucie Unit No. 2
4. Management Controls on St. Lucie Unit No. 2
5. FPL Initiative in Management of St. Lucie Unit No. 2
6. Innovative Construction Techniques
7. Significant Construction Milestones
8. Budget Increase Evaluation
9. St. Lucie Unit No. 1 vs. Unit No. 2 - Milestone Schedule
10. Schedule Comparison of Other Utilities
11. Information Provided to Mr. J. L. Howard

SCOPE COMPARISON

Commodity	St. Lucie Unit No. 1	St. Lucie Unit No. 2	Percent Change
Concrete (CY)	116,320	141,300	21
Formwork (SF)	1,258,000	1,676,020	33
Reinforcing Steel (lb)	28,127,410	28,135,429	-
Embedded parts (lb)	2,425,800	3,983,017	64
Main Steel (Tn)	2,105	2,993	42
Miscellaneous Steel (Tn)	453	548	21
Conduit (LF)	140,000	419,400	200
Duct (LF)	180,000	490,900	173
Cable Tray (LF)	40,000	41,700	4
Power Cable (LF)	540,000	565,000	5
Control Cable (LF)	3,300,000	3,643,000	10
Ground Cable (LF)	150,000	193,000	29
Piping 2" and Under (LF)	177,000	216,800	22
Piping 2½" and Above (LF)	79,400	80,300	1
Valves, for piping, 2½" and Above (Ea)	1,000	1,300	30
Welds, for piping, 2½" and Above (Ea)	5,700	9,000	58

FPL Witness: J. W. Williams, Jr.
Exhibit No.____Document No. 1
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ST. LUCIE UNIT No. 2
CHRONOLOGY OF LICENSING PROBLEMS
AND RESULTING CONSTRUCTION DELAYS

	Elapsed Time	
May 14, 1973	0 mo.	Application formally filed with the Atomic Energy Commission (AEC) to construct and operate St. Lucie Unit No. 2
July 1, 1973	2 mo.	Florida Legislature enacted the Florida Power Plant Siting Act (F.S. 403.501) to be effective October 1, 1973.
September 4, 1973	4 mo.	Complete Application docketed after Environmental Report is accepted by AEC.
January 23, 1974	8 mo.	Pursuant to the State of Florida's Electric Power Plant Siting Act, an application for State Site Certification was filed with the Department of Pollution Control (DPC).
July 24, 1974	14 mo.	Pollution Control Board adopted favorable findings and recommended Order of Hearing Officer on above matters.
October 16, 1974	17 mo.	Environmental and Site suitability hearing begins before Atomic Safety and Licensing Board (ASLB).
February 28, 1975	21 mo.	Partial Initial Decision on environmental and site suitability matters issued by the Nuclear Regulatory Commission (NRC) successor agency to the AEC, ASLB. The order directed the NRC Staff to issue a Limited Work Authorization (LWA), allowing certain work to be undertaken pending receipt of a full construction permit (CP).
March 17, 1975	22 mo.	The LWA was issued by the NRC staff, but could not be used due to delay in receiving certification of the site from the State of Florida. An application for State Site Certification had been filed on January 23, 1974, and although state law mandated a maximum 14 month time period for certification decision, it was not granted until May 18, 1976. (See May 13, 1975 through May 18, 1976.)

	Elapsed Time	
May 13, 1975	24 mo.	Order of State Hearing Officer ruling that issues of radiological health and safety, also considered by the NRC in the construction permit proceeding and previously raised by other parties (DPC, FDER, and others), were not relevant to the proceedings under the doctrine of pre-emption, <u>Northern States Power vs. Minnesota</u>
~~[illegible]~~		
June 16, to July 16, 1975	25 mo.	State Site Certification hearing. Radiological health and safety issues were excluded in accordance with previous order.
July 1 1975	26 mo	[illegible]
July 3, 1975	26 mo.	Intervenors, file appeal of Partial Initial Decision to the NRC's Atomic Safety and Licensing Appeal Board (ASLAB) after two extensions were granted.
October 8, 1975	29 mo.	State Hearing Office filed Findings of Fact, Conclusions of Law and Recommended Order.
December 16, 1975	31 mo.	Governor and Cabinet deny State Site Certification Application. Remanded to Hearing Examiner for additional hearings on radiological health and safety.
February 23-25, 1976	33 mo.	State Site Certification hearing considering radiological health and safety matters.
April 8, 1976	35 mo.	State Hearing Officer's Supplemental Findings of Fact, Conclusions of Law, and Recommended Order.
May 18, 1976	36 mo.	State Site Certification granted by Governor and Cabinet.
June 4, 1976	37 mo.	Limited construction at St. Lucie begins in accordance with the LWA.
June 29, 1976		NRC Appeal (ASLAB) Board issues Order remanding alternative sites contention back to Licensing Board for further hearings to allow Intervenors to cross-examine NRC Staff regarding their 1974 analysis of the issue. LWA allowed to remain in effect with one member dissenting.

	Elapsed Time	
July 28, 1976	38 mo.	Intervenors file motion with NRC Atomic Safety and Licensing Board (ASLB) to reopen "Need for Power" contention.
August 2, 1976	39 mo.	Intervenors file petition for review of the Partial Initial Decision (PID) with the U.S. Court of Appeals for the Washington, D.C. Circuit.
August 13, 1976	39 mo.	Intervenors file Motion for Summary Reversal of the PID and other injunctive relief.
October 21, 1976	41 mo.	U.S. Court of Appeals denies Intervenors motion for summary reversal of the Partial Initial Decision, but decides to stay the LWA. By subsequent order, FPL was allowed until November 8, 1976, to stop all construction activity in an orderly manner.
November 8, 1976	42 mo.	FPL ceases all construction activity on St. Lucie Unit No. 2 pursuant to Court of Appeals order.
December, 1976	43 mo.	Seven days of hearing before the ASLB on alternate sites for St. Lucie 2 and Need for Power.
April 19, 1977	47 mo.	Initial Decision by ASLB issued authorizing Construction Permit (CP).
April - May 1977		Various Motions for Stay of the effectiveness of the CP were filed by Intervenors before NRC and Court of Appeals. FPL responded to Motions with supporting affidavit depicting costs of stay. Motions were denied.
May 2, 1977	48 mo.	CP issued by NRC.
May 12, 1977		Court of Appeals issued Order dissolving October 21, 1976 stay of construction.
June 1, 1977	49 mo.	FPL resumes construction of St. Lucie Unit No. 2.

REGULATORY IMPACT ON ST. LUCIE UNIT NO. 2

Since October 1972, over 1,000 formal NRC requirements have been issued to nuclear plant licensees. These requirements include Regulatory Guides, Standard Review Plans, Bulletins, Circulars, Information Notices, NUREG's, the Code of Federal Regulations (10CFR) and NRC Generic Letters. The requirements provide for new design criteria requirements, analytical evaluations, inspections, etc. The attached chart graphically depicts an overview of the cumulative number of regulatory requirements by year during the St. Lucie 2 design period. It should be noted that other NRC requirements transmitted via NRC correspondence to FPL, NRC questions during the Safety Analysis Reviews, and the verbal requests for additional information are not included in the above.

Although the impact of some of the documents discussed above may be small on an individual basis, the summation of their impact is major, and the resultant cost increases have been significant in terms of direct plant costs and indirectly due to increased complexity in construction.

Many significant regulatory requirements imposed through the years have resulted from events beyond FPL's control (Brown's Ferry, Three Mile Island). As an example, shown below is a chronology of the constant reevaluations of criteria for fire protection systems and the resultant plant impacts. The cumulative impact of these reevaluations is frequently called the "ratchet" effect.

Fire Protection

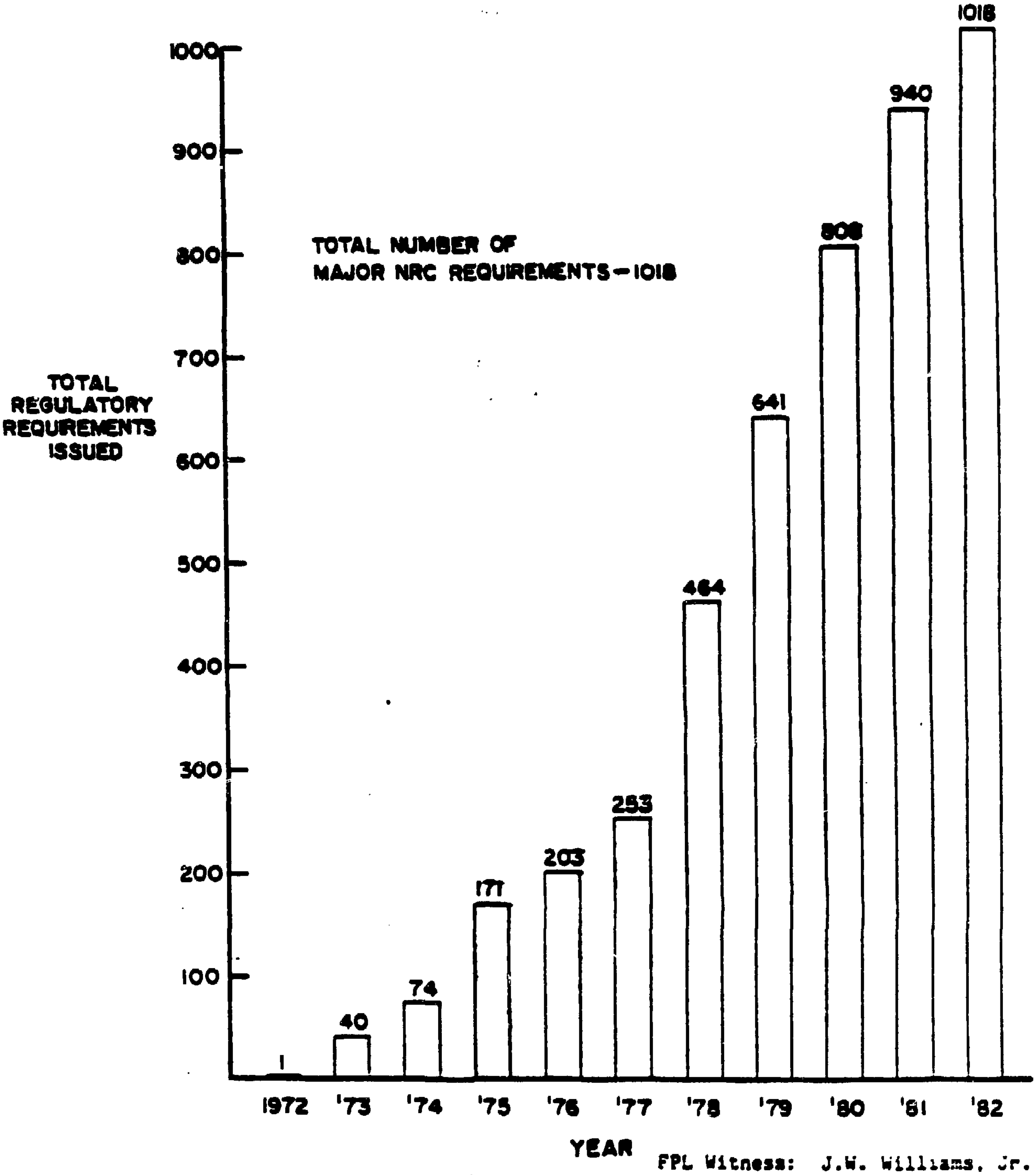
Requirements Issued	Date	Description of Plant Impact
SRP 9.5.1 APCS 9.5.1 App A RG 1.120 RG 1.75	5/76 5/76 7/76 1977 1/78	Summary - A comprehensive fire hazard analysis was initiated which identified potential fire hazards in all areas, postulates credible fires and evaluates the effects of these postulated fires on the operability of systems required to safely shutdown the plant and control the release of radioactivity. As a result of the complete fire hazards analysis, equipment was required, by the NRC, to be added. Some examples are fire seals, hose stations, fire extinguishers, smoke detectors and emergency lighting.
10CFR50 App R	10/78	Appendix R, to 10CFR50, required the applicant to assume a fire has been established. The applicant must now prove that a fire that is contained in a qualified 3hr fire barrier will not damage redundant trains. As a result of this idealogy the following was added: 1) Stairway Enclosures 2) Raise Charging Pump Cubicle Walls 3) Cable Loft Barrier Tray Riser Enclosure 4) A DC Equipment Enclosure 5) Swgr/Cable Spread Room Wall 6) Pressurizer Heater Switchgear Room 7) RCB Electrical Penetration Barrier 8) Cable Tray Bottoms 9) Hatch Covers 10) Diesel Generator Building and Reactor Auxiliary Building Sprinkler System 11) Cable Reroute/Wrap 12) Fire Dampers and Position Switches Energy

Other examples of NRC "ratchets" that have increased plant cost are:

1) Containment Sump Redesign
2) Low Temperature Overpressure Protection (LTOP)
3) Three Mile Island (TMI) Related Items
4) As-Built Piping Review
5) Additional Startup Transformer Switchyard Modifications
6) Physical Independence of Electrical Systems
7) Seismic Design Criteria/Qualification
8) Missile Protection
9) Main Stream Line Break Analyses
10) Pipe Rupture/Jet Impirgement/IEALA
11) Equipment Qualification
12) Quality Assurance

FLORIDA POWER AND LIGHT COMPANY
ST. LUCIE UNIT 2

REGULATORY IMPACT
TOTAL REGULATORY REQUIREMENTS ISSUED VS TIME



FPL Witness: J.W. Williams, Jr.
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MANAGEMENT CONTROLS ON PSL-2

FPL employed a strong Project Management approach in order to achieve its cost, schedule and quality performance objectives during the construction of St. Lucie Unit No. 2. All of the project functions were carried out by a project organization team assigned to the Project General Manager (PGM). The PGM reports to a Director of Projects who reports to the Vice President for Engineering, Projects and Construction. Figure 1 shows the Project Organization Chart and the major project team members. The utilization of a Project Control Supervisor helped to ensure integration of all cost and schedule information to the PGM for use in timely decision-making. FPL personnel were assigned to all major project control activities. They included Engineering, Construction, Purchasing, Contracts, Licensing, Startup, Quality Control, Security, Quality Assurance and Cost and Schedule.

FPL's commitment to strong Project Management and utility control was most evident at the St. Lucie Site. FPL assumed full responsibility for all construction activities utilizing an integrated site organization comprised of both utility and contractor personnel with a single owner site manager, as shown in Figure 2. FPL personnel were assigned to all of the lead support services positions while Ebasco personnel directed the Field Engineering and construction activities. The combined use of an integrated organization and integrated project plan enabled FPL to be more effective in eliciting the best possible performance from the site contractors. FPL personnel received and verified all construction activities, invoices, payroll costs and contract scope changes. Contract administrators were assigned to all major site contractors.

The Project Management controls on St. Lucie Unit No. 2 included numerous systems and reports designed to provide information on virtually every project activity that could impact meeting budget and schedule commitments made to FPL upper management. The primary management baseline control tools included the project integrated schedule and the current approved budget. Monthly cost and schedule status was reported to the utility management throughout the project lifetime in the FPL Project Management Reporting System (PMRS). The master schedule on St. Lucie Unit No. 2 was adopted in March 1977 and maintained throughout the job through the use of a detailed Critical Path Method (CPM) network which contained over 30,000 project activities. Project monthly reports were issued by Ebasco on the engineering services and by the site on the construction and startup activities.

Figure 3 lists the major site control tools that were used to monitor construction productivity and performance. These tools were augmented by daily meetings at the site to ensure the timely identification and resolution of problems. The effectiveness of these tools can be measured by the exceptional performance history of the St. Lucie Unit No. 2 project, which exceeded all other recent nuclear plants by a considerable margin.

In addition to those control tools employed by FPL, each of the major contractors had their own project cost and schedule reporting systems used to ensure this performance to the overall project objectives.

Figure 1

FPL Witness: J. W. Williams, Jr.
Exhibit No. ____ Document No. 4
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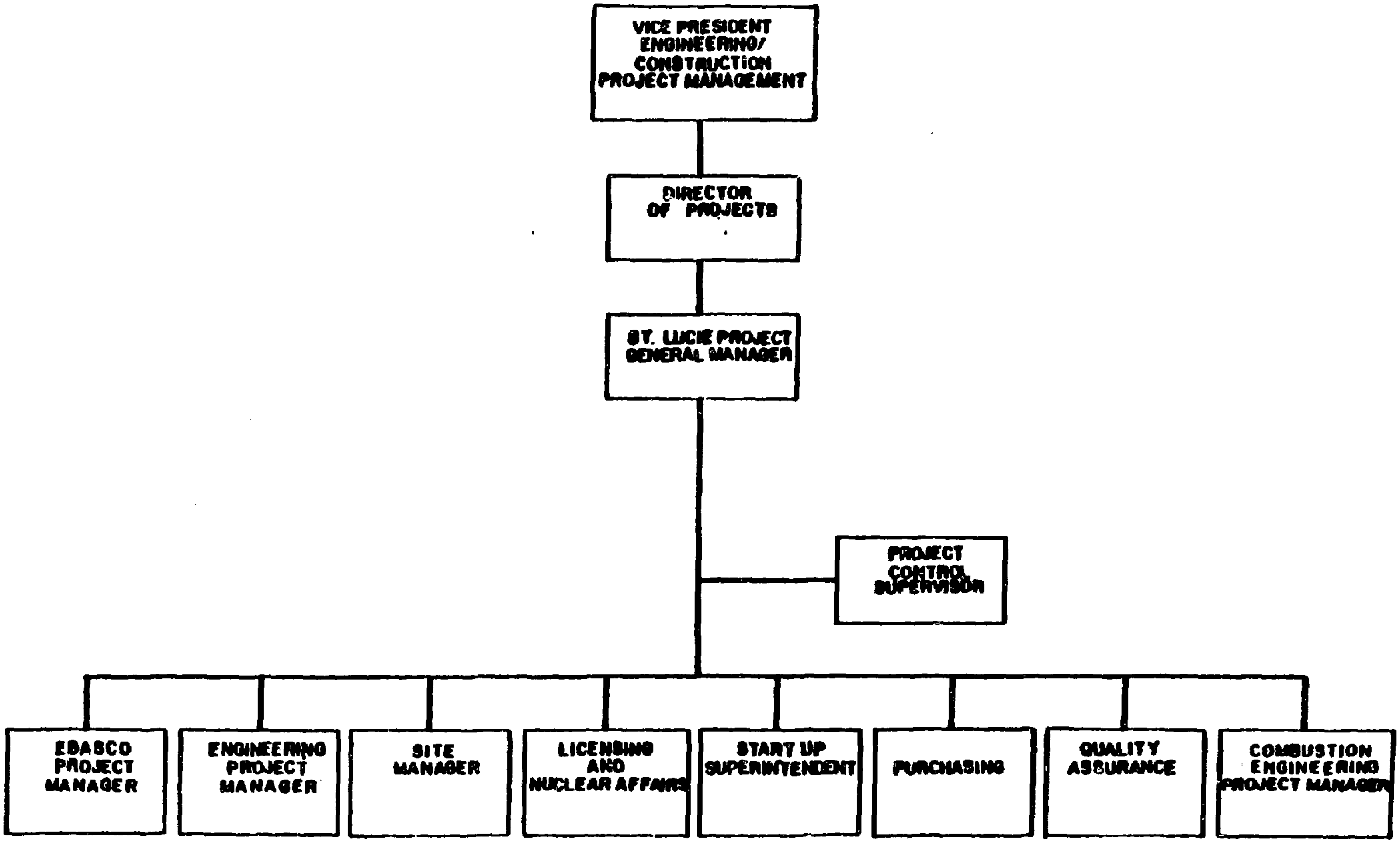
PROJECT ORGANIZATION CHART



ST. LUCIE UNIT NO. 2
SITE ORGANIZATION CHART

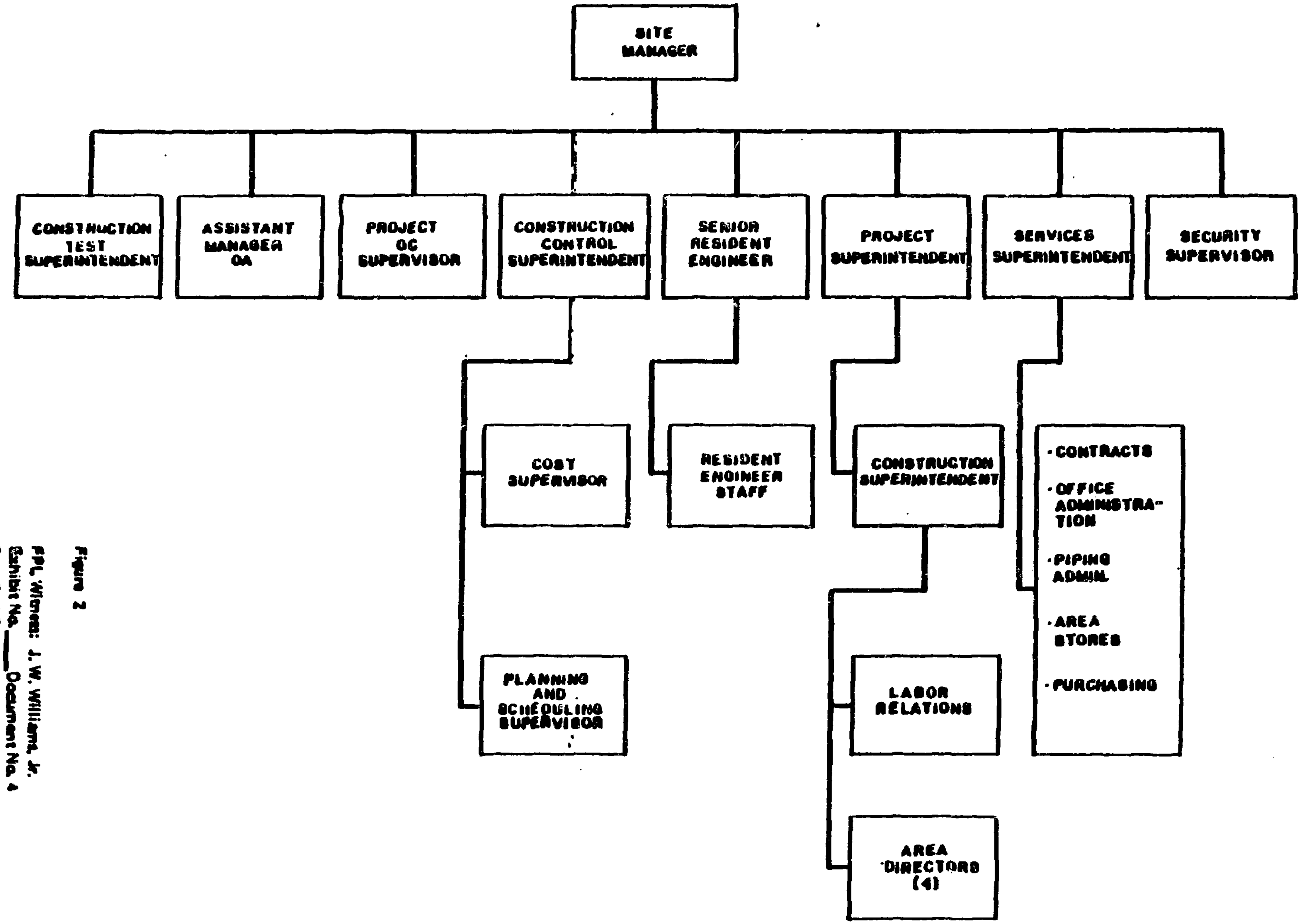


Figure 2

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FIGURE 3

MAJOR PROJECT CONTROL TOOLS USED ON ST. LUCIE UNIT NO. 2

Control Tool	Description
1. Project Master Schedule	Shows top level Project Milestones and Activities
2 Level II/III Schedule	Computerized CPM Network - Construction
3. Startup Schedule	Computerized CPM Network - Startup
4. Schedule Interface System	Shows Cost Account to schedule cross-reference
5. Resource Loaded Schedule	Shows levelized quantities and manhours per schedule
6. Physical Accomplishment Curves	Shows area and project quantity and manhour tracking
7. Productivity Curves	Shows area & project productivity based on actual vs. estimate
8. Material Tracking System	Computerized tracking of all materials on job
9. Electrical Management System	Computerized tracking of all cable, conduit and terminations
10. Project Quantity and Manhour Report	Production and productivity report for each job cost account
11. Bulk Commodity Curves	Shows scheduled vs. actual installation rates
12. System Turnover	Computerized tracking by plant system
13. SCAT	Startup Controlled Accelerated Turnovers allowed for partial system turnover in order to accelerate the startup and testing program.
14. Budget and Cash Flow System	Systems to explain all cost variances by cost category and project category
15. Reforecasting Program	Semi-annual cost report tied to the definitive cost estimate (1977)

FPL Witness: J. W. Williams, Jr.
Exhibit No._____Document No. 4
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16. Construction Report	Monthly report presenting results of all control system indicators
17. Risk Analysis	Program used to estimate probability of meeting schedule and budget
18. Trend Program	Monthly tracking of scope and cost evaluation
19. PMRS	Monthly report to upper management on progress, budget and trends for all projects
20. Management Presentations	Formal semi-annual project progress review presentations to upper management

FPL INITIATIVES IN MANAGEMENT OF ST. LUCIE UNIT NO. 2

The entire history of the St. Lucie Unit No. 2 Project has been one of implementing management action directed toward overcoming obstacles and instilling a sense of urgency in the project team. Literally hundreds of specific actions have been taken by FPL and its contractors to improve productivity, recover schedule time and correct quality concerns. These efforts can best be illustrated by identifying the following five major initiatives taken by FPL which have had the greatest impact on schedule performance.

1) The decision to complete engineering and material purchasing during the Limited Work Authorization (LWA) shutdown period. This reduced the number and severty of schedule impacts which could have resulted from incomplete engineering and material delays.

2) The integration under FPL management of the FPL/Ebasco/Subcontractor site organization and operating procedures. This approach aided in the timely identification and resolution of project problems and concentrated overall responsibility under FPL personnel.

3) The use of a single integrated Project plan, developed early in the job, with <u>total</u> management commitment to meeting the schedule for all major milestones. The benefits of a total commitment to meet all major milestones

was evident in actions taken to overcome impacts on construction of having an operating unit next to it, labor disputes, Hurricane David damage, potential delays in setting the reactor vessel and steam generators, contractor changes in the piping area. It was also evident in FPL's efforts to expedite delivery of orders from many of the equipment suppliers.

4) Early initiation of FPL's system startup program which included development of start-up procedures, refined start-up schedules, identifications of start-up system turnover of partially completed systems to operations personnel and prioritization of construction activities to support system completion.

5) Lastly, FPL demonstrated considerable initiative in expediting the NRC's review of our application for an operating license on PSL-2. This effort included submittal of a letter from the entire Florida State Delegation to the NRC Chairman expressing the need for timely issuance of an operating license, and establishment of a temporary office near the NRC to facilitate a closer working relationship.

Each of the above efforts was undertaken by FPL in order to construct and start up St. Lucie Unit No. 2 in the shortest possible schedule, thereby directly reducing costs.

CONSTRUCTION TECHNIQUES

The optimization of the construction effort was the result, to a large degree, of the early planning and innovative thinking that went into the formulation of the overall construction plan and schedule for this project. The following are some examples of this:

1. Reactor Auxiliary Building "Stair Stepping" Concept

   One of the ideas that went into the initial plan and schedule was the "stair stepping" concept for the construction of the reactor auxiliary building. In this plan, the building was constructed with emphasis placed on early completion of the west end of the building. The benefit of this approach was that early completion of the west end of the structure would provide an early start on the installation of the more critical types of equipment in the building, such as the control room and the reactor auxiliary control boards, the cable vault area, and NSSS auxiliary equipment.

2. Slipforming on Reactor Containment Building

Another innovative construction approach at St. Lucie Unit No. 2 was the "slipforming" of the concrete containment shield wall for the reactor containment building, in lieu of the traditional "step-form" method. "Slipforming" is a continuous concrete casting process which involves the steady movement of a single set of forms up a concrete structure until the entire structure is complete. The shield wall is a three foot thick concrete cylinder which is approximately 190 feet high with an inside radius of 7 4 feet. It is supported by a reinforced concrete ring wall (9 feet thick and 4 feet high) which, in turn, rests on the reinforced concrete base mat. The shield wall contains more than 1,000 tons of reinforcing steel with another 23 tons of embedded materials such as electrical conduits, grounding cables and anchor bolts.

Shield wall placement through slipforming of 10,000 cubic yards of concrete averaged vertical 11½ feet per day, and the operation took place without interruption in only 16½ days in November, 1977. Manpower for slipforming averaged 398 craft workers, and the crafts worked three shifts a day, seven days a week until completion. Immediately after completion of slipforming, construction on the steel containment liner was able to start inside the shield wall. Constructing the shield wall using the traditional "step-form" approach would have taken 98 days and utilized more than double the manhours.

3. Nuclear Steam Supply System (NSSS) Installation

An important benchmark in the NRC's assessment of nuclear plant construction is the installation of the nuclear steam supply system major equipment, i.e., the reactor vessel, steam generators and pressurizer. We were able to reach this milestone on a very short schedule by adopting two innovative approaches.

The first of these innovations was the design of the steel containment liner to utilize a "tops-off" approach, together with the early planning necessary to allow the use of that approach. Basically, this method allowed the steel walls of the containment to be heat treated when it was complete while at the same time leaving the top of the structure open. As a result of using the "tops-off" approach, interior concrete work started months earlier than otherwise would have been possible and ensured that support structures were ready for NSSS installation.

Secondly, a system of temporary bracing was used to support the polar crane while the reactor vessel was being placed inside the containment building. By using this temporary bracing it was possible to place the vessel without waiting until the interior concrete was brought up to the operating level, as would usually be the case. Thus saved considerable schedule time and enabled construction forces to meet the target date of June 1980 for setting the reactor vessel.

4. Jobsite Labor Relations

Specific labor relations programs instituted include the following:

a) Quarterly labor-management meetings designed to open lines of communication in a non-adversary atmosphere and provide a means of informing the building trades as a group on upcoming project manpower needs. It also provided a means of resolving grievances and jurisdictional disputes and generally improving the labor-management climate.

b) Special training was given electrical foremen. This was supervisory-type training, and included a formal course concentrating on how improved planning and work process analysis by foremen can lead to major productivity gains. Electricians also received training in the methods used at St. Lucie Unit No. 2 in cable spooling and cable terminations; this has resulted in reasonably smooth operation in these areas. In addition to the above, welding training was made available to all appropriate crafts.

Enlightened and prudent management techniques recognize that management cannot "drive" a work force--it must lead and motivate the work force to reach achievable goals. We believe that the St. Lucie Unit No. 2 management team has effectively employed this leadership philosophy in the labor relations at the project.

5. Safety

Jobsite accidents have a costly impact on the $300 billion-a-year United States construction industry. Work-related injuries and illnesses, including fatalities, occur in construction at a rate that is 54% higher than the rate for all industries, making it one of the most hazardous occupations.

FPL has tried aggressively to improve on this record at St. Lucie Unit No. 2.

The safety program in effect at the St. Lucie Unit No. 2 site has resulted in receipt of two safety awards for 1982 for working over one million manhours <u>without</u> a lost time accident. In 1981, the site O.S.H.A. index was 98.29 percent better than the Bureau of Labor Statistics average. This performance has resulted in significant direct and indirect savings to FPL by reducing insurance and accident claims, and by avoiding the loss of productivity and schedule disruptions which result from accidents.

Florida Power & Light Company
St. Lucie Unit No. 2 Plant in Service*

(000)

Line No.			(1) Beginning Balance	(2) Net Additions	(3) Ending Balance
1	June	1983	$ -0-	$1,189,827	$1,189,827
2	July	1983	1,189,827	6,285	1,196,112
3	August	1983	1,196,112	2,576	1,198,688
4	September	1983	1,198,688	1,948	1,200,636
5	October	1983	1,200,636	2,124	1,202,760
6	November	1983	1,202,760	1,313	1,204,073
7	December	1983	1,204,073	2,639	1,206,712
8	January	1984	1,206,712	1,958	1,208,670
9	February	1984	1,208,670	717	1,209,387
10	March	1984	1,209,387	3,903	1,213,290
11	April	1984	1,213,290	538	1,213,828
12	May	1984	1,213,828	444	1,214,272
13	June	1984	1,214,272	3,278	1,217,551
14					
15	13 Month Average				$1,205,831

*Adjusted to reflect only FPL's 85.10449% ownership share in the Unit.

Columns may not add due to rounding.

FPL Witness: J. W. Williams, Jr.
Exhibit No. _____ Document No. 11
Page 2 of 4 (January 31, 1983)

Florida Power & Light Company
St. Lucie Unit No. 2 - O&M Expenses
($000)

Line No.		Total	July 1983	August 1983	September 1983	October 1983	November 1983	December 1983	January 1984	February 1984	March 1984	April 1984	May 1984	June 1984
	O&M Costs													
1	Payroll	$ 6,787	$ 490	$ 490	$ 482	$ 648	$ 479	$ 483	$ 614	$ 619	$ 619	$ 620	$ 621	$ 622
2	Contractor	785	58	58	59	59	59	59	72	72	72	72	72	73
3	Materials & Supplies	700	56	56	57	57	57	57	60	60	60	60	60	60
4	Other	5,091	448	448	448	448	449	450	400	400	400	400	400	400
5	Payroll Loading	1,554	112	112	110	148	110	111	141	142	142	142	142	142
6	Less O&M Included in Base Case	(401)	(33)	(33)	(33)	(33)	(33)	(33)	(33)	(34)	(34)	(34)	(34)	(34)
7	Total O&M Before Credits	14,516	1,131	1,131	1,123	1,327	1,121	1,127	1,254	1,259	1,259	1,260	1,261	1,263
	O&M Credits(a)													
8	Participation Credits	(3,764)	(288)	(288)	(296)	(329)	(294)	(296)	(345)	(322)	(337)	(317)	(318)	(334)
9	Credit Included in Base Case	39	3	3	3	3	3	3	3	3	3	4	4	4
10	Total Credits	(3,725)	(285)	(285)	(293)	(326)	(291)	(293)	(342)	(319)	(334)	(313)	(314)	(330)
11	Net O&M Expenses	$10,791	$ 846	$ 846	$ 830	$1,001	$ 830	$ 834	$ 912	$ 940	$ 925	$ 947	$ 947	$ 933

(a) Represents reimbursement of cost by Orlando Utilities Commission and Florida Municipal Power Agency (FMPA) for their share of ownership of St. Lucie Unit No. 2.

FPL Witness: J.W. Williams, Jr.
Exhibit No. ___ Document No. 11
Page 4 of 4 (January 31, 1983)

BUDGET INCREASE EVALUATION

DESCRIPTION	ORIGINAL BUDGET ($000)	CHANGES 1973 - 1983: REGULATION ($000)	FPL INITIATED SCOPE CHANGES ($000)	ESCALATION CHANGES ($000)	DESIGN/ ESTIMATE REFINEMENT ($000)	TOTAL CHANGES ($000)	TOTAL BUDGET ($000)
Nuclear Steam Supply System	$ 46,390	$ 57,951	$ 11,080	$ 4,694	$ 2,325	$ 76,050	$ 1[illegible]40
Turbine Generator	28,810	0	5,084	0	(1,493)	3,591	[illegible],401
Other Directs	168,600	146,726	27,118	47,461	71,174	292,479	461,079
A/E Services	17,500	69,340	16,876	265	60,398	146,879	164,379
Indirects	32,700	186,670	29,553	35,347	69,820	321,390 ~~320,390~~	354,090
AFUDC	66,000	123,941	18,643	34,273	42,754	219,611	285,611
TOTAL	$ 360,000	$584,628	$108,354	$122,040	$244,978	$1,060,000	$1,420,000
		55.2%	10.2%	11.5%	23.1%	100%	

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ST. LUCIE UNIT NO 2
SCHEDULE COMPARISON
(SOURCE NRC YELLOW BOOK)

UNIT	FUEL LOAD	
MCGUIRE I	1/81	117
LASALLE I	4/82	103
GRAND GULF I	5/82	92
SUSQUEHANNA	8/82	100
SUMMER I	8/82	112
SHOREHAM I	2/83	124
SAN ONOFRE 2	2/82	96
WATERFORD 3	5/83	102
ST. LUCIE 2	3/83	71
BYRON I	8/83	100
ENRICO FERMI 2	6/83	169
COMANCHE PEAK I	6/83	104
CALLAWAY I	4/84	103
MIDLAND 2	7/83	124
WATTS BAR I	8/83	127
PALO VERDE I	8/83	87
WASH. NUCLEAR 2	9/83	133
PERRY I	11/83	109
SEABROOK I	9/84	99
WOLF CREEK I	10/84	93
LIMERICK I	10/84	173
CATAWBA I	10/84	125
HARRIS I	12/84	131
BRAIDWOOD I	4/85	116
RIVER BEND I	4/85	72
BELLEFONTE I	5/85	128
WASH. NUCLEAR 3	6/85	98
MILLSTONE 3	12/85	139
BEAVER VALLEY 2	12/85	140

29 PLANTS

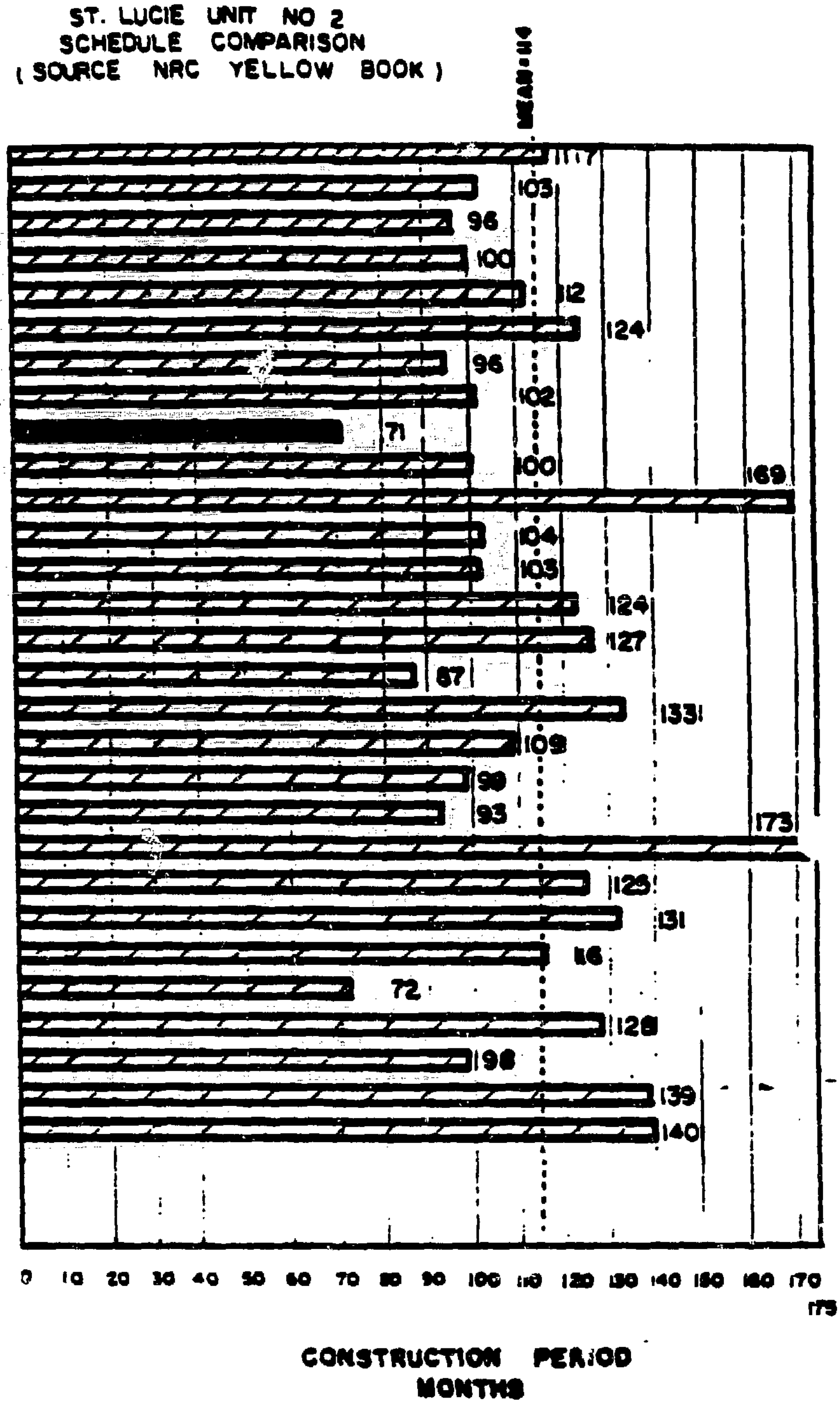

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Exhibit No. ___ Document No. 8
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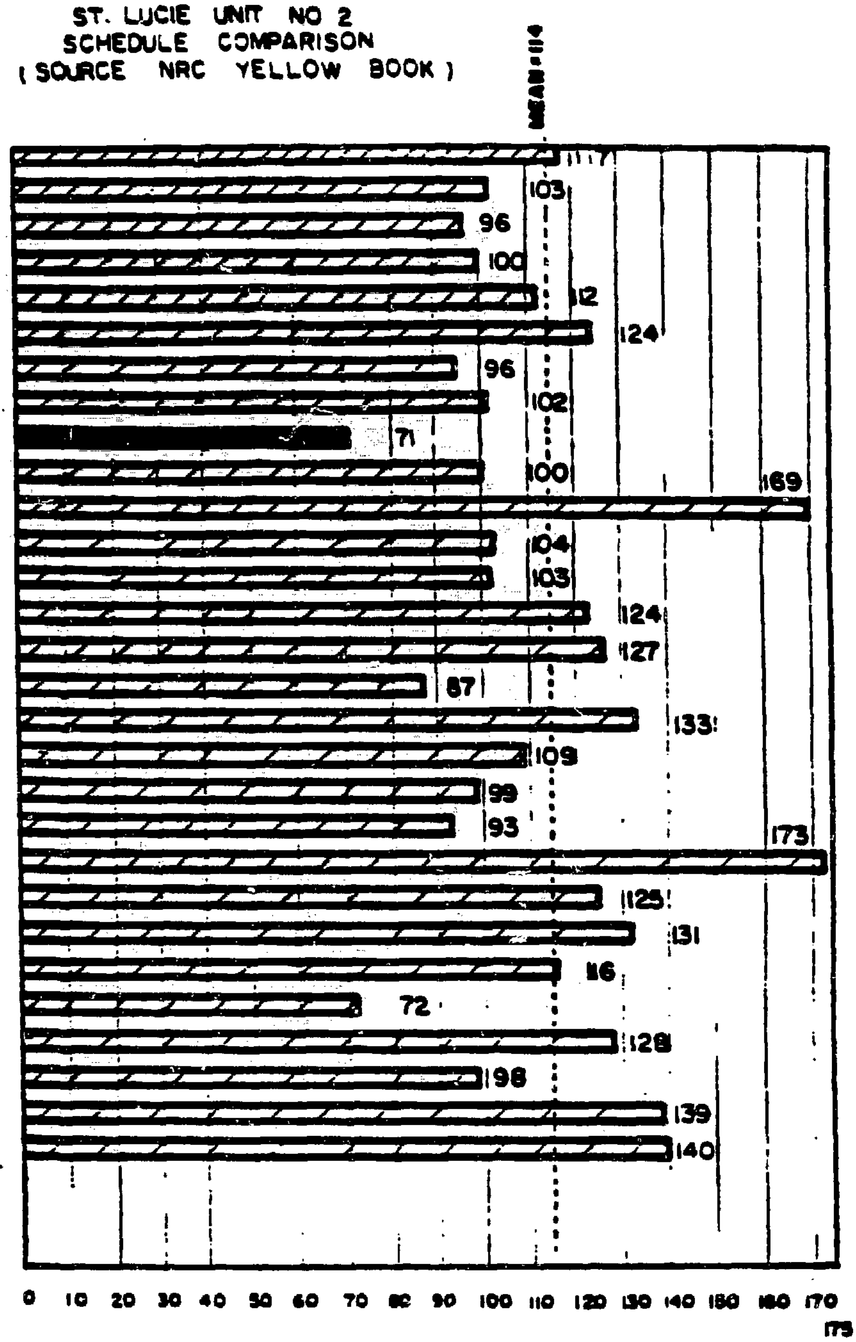
29 PLANTS



CONSTRUCTION PERIOD
MONTHS

FPL Witness: J.W. Williams, Jr.
Exhibit No. ____ Document No. 10
Page 1 of 1 (January 31, 1983)

Florida Power & Light Company
St. Lucie Unit No. 2 Plant in Service*

(000)

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($000)

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(a) Represents reimbursement of cost by Orlando Utilities Commission and Florida Municipal Power Agency (FMPA) for their share of ownership of St. Lucie Unit No. 2.

III NUCLEAR RELIABILITY IMPROVEMENT & SAFETY OPERATIONS

W. G. Counsil
Executive Vice President
Texas Utilities Generating Company

NUCLEAR RELIABILITY IMPROVEMENT & SAFETY OPERATIONS

W. G. Counsil
Executive Vice President
Texas Utilities Generating Company

I. Introduction

Since most of my eighteen years of commercial nuclear experience has been at Northeast Utilities in New England, I will discuss that program in some detail when talking about reliability improvement and safety of nuclear operations. In addition, if one were to be placed in an ideal position of having the hindsight of thirty years of worldwide commercial experience already like many of you, how should one go about starting into nuclear power. Also, I will discuss the reliability of plants against catastrophic failure as determined by probabilistic risk assessment (PRA) methods and against less severe failures as measured by capacity factor. I will present Northeast Utilities (NU) plant capacity factor data for the past eight years and discuss how it compares to worldwide figures. The reasons for variations in NU's year-to-year plant reliability will be explained. I will discuss briefly the organizations which have been developed in-house that perform reliability functions and their general area of expertise.

I will then talk about several of the specific reliability programs underway at NU. These include equipment vibratory analysis, PRA failure studies, the selection process for large spare components, design reviews for maintainability, root cause failure analysis and NU's program for review of industry experience. This will be followed with concerns I have on the impact of Nuclear Regulatory Commission (NRC) backfits on plant reliability and safety. I will make some comments on the activities of industry groups such as INPO and the various Owner Groups. I will conclude this portion with a look at the future and point out events that I believe will improve plant reliability and those which will harm it.

Finally, I will review my personal insight into nuclear safety; that nuclear safety is not a collection of codes and criteria. It is truly an ethic that must pervade an entire organization. It encompasses supervisory responsibilities as managers, leaders and, most importantly, trainers of their people. It requires strict adherence to procedures and an especial alertness when modifications are made to the plant and its procedures.

I feel strongly that the path to safer, more reliable nuclear plants lies with the industry working as a team, sharing the experiences learned both good and bad. A logical vehicle for this teamwork is an industry organization such as the Institute of Nuclear Power Operation (INPO).

II. Background

NU currently operates three nuclear power plants, each purchased from a different reactor supplier. Connecticut Yankee, a four-loop Westinghouse PWR that went commercial in January, 1968; Millstone Unit No. 1, a General Electric BWR that went commercial in December, 1970; and Millstone Unit No. 2, a two-loop Combustion Engineering PWR that went commercial in December, 1975. In addition, NU is currently constructing Millstone Unit No. 3, a four-loop Westinghouse PWR that is scheduled for commercial operation in May, 1986. This diversity, although not really planned, does provide a degree of protection against common-mode faults of a particular reactor system.

NU ventured into the nuclear business in the late 1950s as a participant and part owner of Yankee Rowe along with a consortium of other New England utilities. Later, Connecticut Yankee and the other Yankee nuclear power plants were constructed by this group of New England utilities. The Yankee Atomic Electric Company (YAEC) was formed by the Yankee plant owners to provide technical and administrative services to the Yankee plants.

During the mid-1960s, NU moved forward with its own nuclear program with the purchase of Millstone Unit No. 1, on a turnkey basis, from General Electric. NU had minimal responsibility during design and construction due to the terms of the contract. Its greatest involvement was through participation in the licensing process and development of an operating staff.

In 1968, while NU was finalizing the contracts for Millstone Unit No. 2, there was a clear recognition of the need to form an in-house nuclear engineering staff to oversee the increasing commitment to nuclear power. During the fall of 1968, NU formally organized a Nuclear and Mechanical Engineering Department and staffed it initially with 20 engineers and technicians. Limited nuclear expertise was obtained by hiring engineers from reactor vendors and by providing in-house training assignments at Yankee Rowe and YAEC. This initial group served mainly in a review capacity to ensure that proper and current codes and standards were being applied to reactor components, systems and structures. A quality assurance function was established at this time.

Gradually, the Nuclear and Mechanical Engineering Department expanded, taking on certain pieces of technical work that had previously been contracted. In 1971, as a major owner of Connecticut Yankee, NU took over technical responsibility from YAEC for all plant functions. In 1973 NU formally established a Quality Assurance Department and in 1975 a Reliability Department was established.

As their commitment to Nuclear power grew, and the regulatory climate governing engineering, operation and licensing of nuclear power plants became more complex, their organization proved unable to meet the challenge. In 1978, NU reorganized its engineering and nuclear operations function. All nuclear support activities, which include engineering, operation, construction and licensing, were placed under one corporate entity forming the NU Nuclear Engineering and Operations (NE&O) Group. The present NE&O group has a technical staff of about 900 persons exclusive of the nuclear plant staff.

III. Reliability Against Catastrophic Failure

In the nuclear power business, both the utilities and the regulators have an overriding concern that a catastrophic failure will occur. This is generally defined as a sequence of events, which once initiated, leads to a reactor "core melt." The consequences of events after a core melt are not well known and are, therefore, shrouded in considerable uncertainty. To compensate for this uncertainty and to be sure the results are enveloped by the analysis, large measures of conservatism are applied.

However, the events and necessary failures needed to result in a core melt can be defined. Also, the methodology for performing these analyses is well developed. The two prerequisites are people experienced in probabilistic risk assessment (PRA) and people with a detailed working knowledge of the plant, including familiarity with control logic, procedures and testing intervals.

Several utilities in the United States have performed PRAs for their nuclear plants, particularly those near large population centers where the risks from a core melt accident are relatively higher. NU completed a comprehensive probabilistic risk assessment (PRA) of Millstone Unit No. 3 (MP-3). The year before, a joint PRA with the NRC on Millstone Unit No. 1 (MP-1) (NUREG/CR-3085) as part of the NRC's Interim Reliability Evaluation Program (IREP) was completed.

The results of these studies are generally reported in terms of probability of core-melt per year of operation. However, the reliability of a system or component is usually defined as one minus the probability of failure, that is

$$\text{Reliability} = (1 - P(\text{fail}))$$

Generally, this formulation works out to be a string of "nines" preceded by a decimal point. In reliability engineering jargon this is frequently referred to by stating the number of "nines". A failure probability of 10^{-5} would result in a "reliability = 0.99999" or "five nines".

Studies that NU has done have shown that MP-1 has a reliability against catastrophic failure, excluding external events, of about 0.9999 or "four nines", and MP-3 has a reliability of about 0.99995 or "four and one-half nines". These values are essentially within the draft NRC guidelines of "four nines" (0.9999) and are believed to be acceptable.

However, of major concern to NU and other utilities, I suspect, is the failure of individual systems or components which, although not catastrophic, cause unexpected or extended plant outages. These frequently expose the utility to adverse publicity through the media and elicit a negative reaction from the regulators. They also cause large expenditures for replacement power. On NU's system where replacement power for nuclear is generated by burning oil, an 870 MWe nuclear unit incurs a charge of 850,000 dollars for every day it is out of service. In addition, there is the cost to repair the failure which can run in the tens of millions of dollars.

IV. NU's Nuclear Plant Reliability

A good measure of the overall reliability of a power plant relative to other power plants is a comparison of "capacity factors". For nuclear plants with about a twelve month refueling cycle that cannot refuel on-line, an upper limit on capacity factor is about 90%, with the expected value being about 70% due to random failures and necessary maintenance outages. Occasionally a nuclear plant will operate a full calendar year without refueling, as Connecticut Yankee did in 1978. This tends to lead to an abnormally high annual average capacity factor for that particular year. Therefore, five to ten-year average capacity factors are usually viewed as more meaningful.

In Table I, I have shown the capacity factors of NU's nuclear plants for the past eight years, that is from 1977 through 1984. Also, Table I shows the 1982 world average nuclear plant capacity factors for BWRs and PWRs. These average out to about 61%. We can also see that the eight-year average of NU's nuclear plants is about 72%, which is significantly above the current world average and in the range one would expect. NU is particularly proud of the reliable operation of Connecticut Yankee. With an eight-year average capacity factor of about 79%, it is one of the world leaders in reliability for off-line refueled plants.

The data in Table I shows considerable variability due to random plant equipment failures and shutdowns for NRC mandated backfit projects. A tabulation of the cause of major non-fueling outages is presented in Table II for each of NU's nuclear plants. When Table II is compared with Table I, it shows the cause of the low value of capacity factor for certain years.

Millstone Unit No. 2 has been plagued with equipment problems during each of the past seven years except for 1981. In 1977 the condenser was

TABLE I

NU NUCLEAR PLANT PERFORMANCE

PLANT	CAPACITY FACTOR, 96								
	1977	1978	1979	1980	1981	1982	1983	1984	8-Year Avg.
Connecticut Yankee	79.7	93.5	81.7	69.9	79.9	89.0	74.2	65.9	79.2
Millstone Unit 1	83.4	80.5	73.0	58.5	43.6	70.5	92.6	74.7	72.1
Millstone Unit 2	59.7	61.9	58.6	63.9	89.9	65.7	32.2	87.7	65.0
NU 8-Year Average									72.1

World Average for 1982*

BWRs	61.2
PWRs	60.0

*From Nuclear Engineering International, Vol. 28, No. 339, April 1983, pps. 39-41.

TABLE II

NU NUCLEAR PLANT RELIABILITY

Plant and Year	Cause of Outage
Millstone Unit 2	
1977	Condenser Retubed
1978	Steam Generator Modifications
1979	FW Pipe Inspection & Repair
1980	Seismic Restraints
1982	RCP Seal Replacement - Steam Generator Tube Inspections
1983	Steam Generator Tube Sleeving Project and Thermal Shield Removal
Millstone Unit 1	
1980	Piping Restraints on Isolation Condenser Turbine Expansion Joints, LPCI Supports
1981	"B" LP Turbine, Removal of 14th Stage
1982	Operation with 14th Stage "B" LP Turbine Removed
Connecticut Yankee	
1980	Turbine Limit, RCP Seal, TG Overspeed
1984	Refueling Pool Seal Failure

retubed; in 1978 the steam generator underwent modifications; in 1979 the feedwater system components required inspection and subsequent weld repairs; in 1980 NRC mandated seismic restraints were installed; and in 1982 the reactor coolant pump seals required replacement. In 1983 several thousand steam generator tubes were sleeved or plugged, the thermal shield was removed, and many fuel assemblies required replacement/repair.

Millstone Unit No. 1 has operated with a reasonable capacity factor over the review period except for 1980 and 1981. In 1980 piping restraints were installed on the isolation condenser and the turbine expansion joints were replaced. In 1981 problems were encountered with the steam turbine, and the 14th stage of the "B" LP turbine was removed, as well as the "A" for balance purposes. This not only caused "downtime", but it lowered the power output until the stage could be replaced some 15 months later.

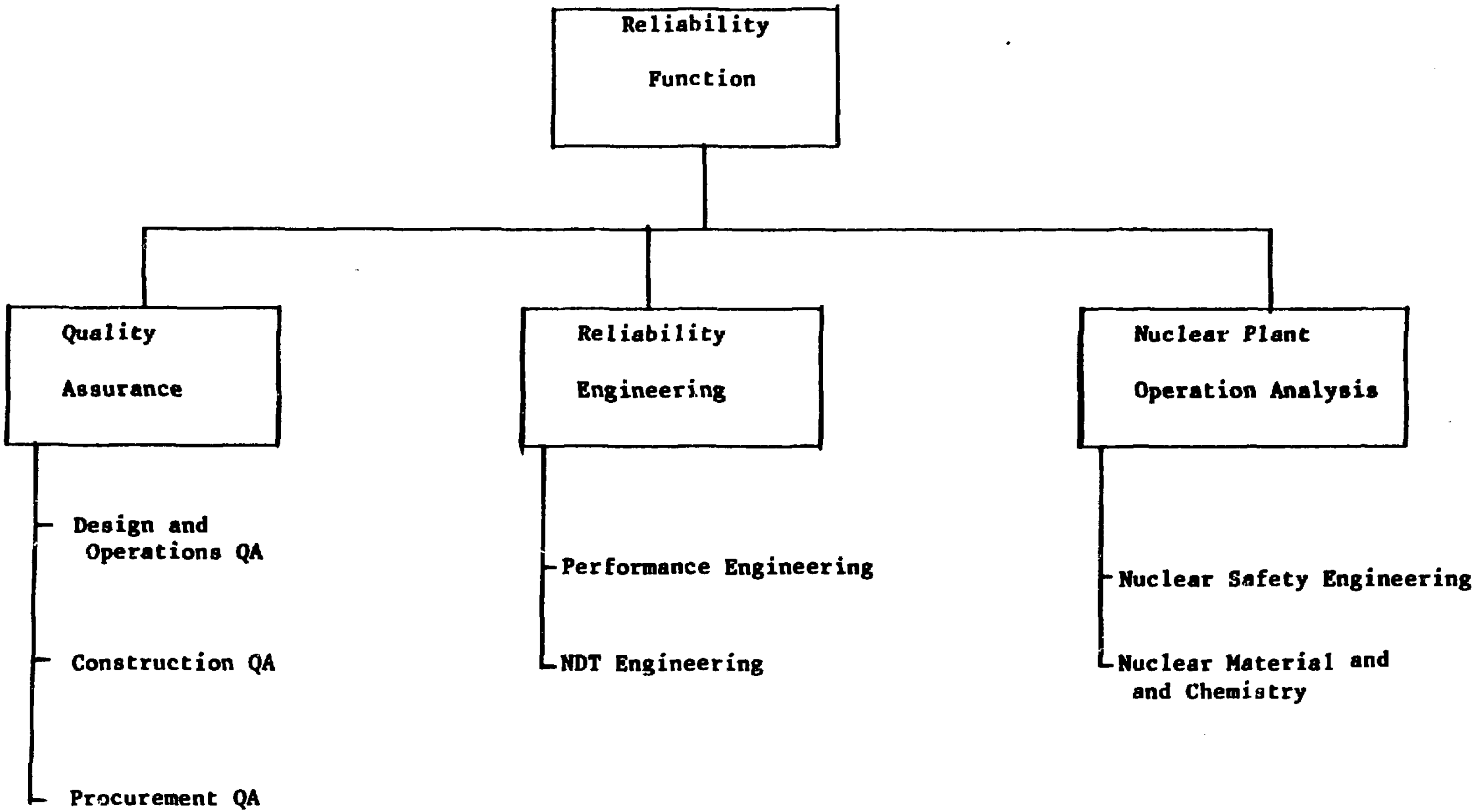
As I mentioned earlier, Connecticut Yankee has been an excellent performer. Its lowest capacity factor was in 1984 when it dropped to about 66% due to the reactor cavity pool seal failure early in the refueling outage.

V. NU Reliability Organization

At NU the reliability function is performed primarily within three branches of the NE&O organization. Of course, reliable plant operation is expected to be foremost in the minds of all of the staff. However, the organizations formally assigned the task of assuring or improving the reliable construction and operation of the nuclear plants are the Quality Assurance Branch, the Reliability Engineering Branch, and the Nuclear Plant Operations Analysis Branch. These three organizations are staffed with a trained cadre of professionals who are equipped to handle the tasks at hand. There are 44 people in Quality Assurance, 39 people in Reliability Engineering, and 33 people in Nuclear Plant Operations Analysis, for a total strength of 116 technical people. These people are all directly involved in programs that support and improve the reliability and quality of NU's nuclear plants. The organization is shown functionally in Figure I.

FIGURE I

NU RELIABILITY FUNCTIONS



The Quality Assurance Branch reviews plant drawings and specifications; audits plant construction, modification and operation; evaluates suppliers prior to placing orders; provides NDE training and certification, and performs manufacturers inspections. All of these activities are directed towards improving the quality and reliable operation of the nuclear plants. Many are required by the NRC.

The Nuclear Plant Operations Analysis Branch monitors the experience of the nuclear industry by reviewing and evaluating Licensee Event Reports (LERs), Plant Operating Events and recent input to the Nuclear Plant Reliability Data System (NPRDS). They also provide material engineering and coolant chemistry requirements. This latter function is particularly important with respect to prolonging the operational life of steam generators in the PWRs.

The Reliability Engineering Branch performs a variety of equipment tests, collects and analyzes operational data, performs vibration analysis and balances rotating equipment, does root cause failure investigations, develops Inservice Inspection (ISI) programs and performs non-destructive examinations and closed-circuit TV inspections, reviews designs for maintainability, performs heat balance calculations and heat loss tests.

These reliability activities are listed in Table III. Many of them have been instrumental in improving the reliability of NU's power plants, both nuclear and conventional

VI. NU's Reliability Program

As you can envision from the size of NU's reliability organization and the scope of their activities, the reliability program has considerable breadth and depth.

A. Equipment Vibratory Analysis

The NU rotating equipment vibratory analysis program provides support for equipment maintenance and the solution of vibration problems such as bearing and other rotor dynamic instabilities.

TABLE III

NU RELIABILITY ACTIVITIES

Reliability Engineering	Quality Assurance	Nuclear Operations Analysis
Collect/Analyze Operating Data	Drawing Review	Evaluate LERs (INPO SEE-IN)
Perform Equipment Testing	NDE Training/Certification	Evaluate NU Plant Operating Events
Heat Balance Calculations	Operating Plant Audits	Review NPRDS
Design Review for Maintainability/ Reliability	Construction Audits	Coolant Chemistry
Root Cause Failure Investigation	Supplier Evaluations	Corrosion Control
Balance Rotating Equipment	Manufacturers Inspections	Material Engineering
Availability Analysis/Improvement	Engineering Specification Review	
ISI Program Development/Implementation		
NDE Inspections		
CCTV Inspections		
Heat Loss Tests		
Integrated Leak Rate Tests		

The program utilizes modern equipment, such as spectrum analyzers, dual channel tracking vector filter analyzers, multiple channel plotters and vibration sensors installed at each bearing location. The test equipment provides the necessary rotor dynamic response information to perform vibration analysis. It has been successfully used for multiplane balancing without trial weights, alignment correction, and signature and trending analyses.

The nuclear Inservice Inspection Report requires monthly vibration monitoring of certain pieces of equipment. As part of this program, overlaying the amplitude vs frequency curves for one of Millstone Unit No. 2 feed pumps permitted NU to detect an incipient bearing failure. Not wanting to shut the unit down at that time, NU increased the inspection frequency, charted the bearing deterioration and was able to perform the bearing replacement during a later convenient scheduled outage.

One of NU's large steam turbine generator units historically required numerous balance trial weights following an overhaul. Using vibration analysis with reference angles and a computer program, it has been possible to add balance weights at six different locations simultaneously. This greatly reduced critical path time by allowing early start-up and continued plant operation.

Vibratory analysis serves as a final quality control check on equipment assembly by comparing the pre and post maintenance signatures. The signature will confirm that bearings have been installed properly and the shaft aligned.

B. Design Review for Maintainability

Maintainability has increased in importance with increasing cost of downtime and the increased complexity of plant components and systems. Adding to the maintenance complexity and the spacial confusion are the new NRC mandated seismic restraints which infringe upon the designated laydown space.

To capture major maintainability problems, NU borrowed from the chemical industry practice of making plastic scale models of the units under construction. The MP-3 model was made concurrent with the layout drawings in an area near the Engineer/Constructor's design staff. The proximity of the model to the designer's drawing tables encouraged verification of layout concepts. It also permitted a quick check on recent design changes, such as major seismic restraints, which may not have been required when the unit was initially designed.

The model has enabled plant maintenance staff to visually "walk through" the major maintenance activities to ensure that the evolutions of disassembly, laydown, repair, move-in and move-out, and reassembly can be accomplished. Walking through a major component removal, such as a long shaft vertical pump, indicates the necessity of hatches and rigging points in a floor above the pump. These considerations permit removing the pump without disassembly and subsequent reassembly outside the compartment, easing the problems associated with re-alignment.

C. Spare Components Selection Process

NU has developed a system, based on cost and unit availability, for aiding in the decision of stocking spare major components whose failure could have a significant impact on unit reliability and availability. These are components such as main turbine rotors and large electric motors. The high cost impact, low failure probability of these components poses difficult questions.

NU developed a quantitative method to develop and prioritize the component critical list based on the particular component's worth to the generating system. The worth to the system is defined as the "increased availability vs cost factor". This factor is determined by combining the component's effect on unit capacity, the probability of component failure, the replacement power costs, the component storage costs, the cost and lead time to manufacture, the cost of capital, and the component replacement/repair time.

Each component on the critical list which has a cost and availability consequence is compared, based on a cumulative revenue requirements difference (CRRD). This CRRD method yields an expected value of the component failure on the customer's cost of electricity. These results provide management with a means to prioritize and control the expenditures of funds for major spare components.

D. Root-Cause Failure Analysis

Root-cause failure analysis has proven to be an effective process for identifying events and parts that are responsible for a component's final failure. The process involves a thorough analysis of the failure process by persons skilled in the application of engineering fundamentals and who understand the systems operation.

NU has organized a team of experienced technical personnel to perform root-cause analysis. The team members approach a problem with objectivity, verifying whether the design is proper for the application and if the component in question is operated according to specifications. If necessary, additional data is acquired by way of test instrumentation to further the analysis.

Two recent examples of problem-solving relate to a feed pump and a hydro unit. At Millstone Unit No. 2, the steam turbine feed pump oil lube design was modified, resulting in a saving of $300,000 and elimination of substantial down-time for reinstallation recommended by the vendor. At the Northfield Mountain pumped storage hydro unit, design modifications were made to correct a generic high bearing temperature problem which the vendor was unable to correct for six years.

E. Improved Operations Reliability/Availability

Because of the significant effect of nuclear generated capacity on system economics, it has become critical to ensure that the nuclear units are operating at maximum efficiency.

NU has developed a program which collects operating plant data, consisting of approximately 100 pressures, temperatures and flows throughout the thermodynamic cycle. The data is collected via the plant process computer and stored on the corporate computer system. The data reduction and analysis process has been computerized and consists of computer generated trend plots, computerized screening of the data, and flagging of abnormal operation and instrumentation calibration drift. The final step compares the data to baseline values. An operational baseline period, indicative of maximum efficiency, is chosen and used for comparative purposes. Expected deviations, due to seasonal changes, power level, etc. are accounted for and automatically adjusted within the software package. A megawatt accountability table is produced for operational/engineering review. With the automatic computer trending and the ability to manipulate data through user friendly software, the thermodynamic analyst is able to locate problem areas.

Concurrent with the analytic capability, NU has developed computerized heat balance models which have been benchmarked with actual component operating data. These programs allow one to evaluate cycle changes caused by backfits and repairs, and to optimize replacement component design. In addition, the program establishes a realistic maximum output goal for the unit.

Through combining the two computerized programs, NU is able to ensure that the units are operating at true maximum expected output for the greatest amount of time possible.

F. PRA Evaluations

NU has been using probabilistic risk assessment (PRA) for evaluating test and surveillance intervals and to ascertain the safety significance of plant backfit projects.

Many of the Technical Specification test and surveillance requirements for nuclear power plants were evolved over a decade ago with only a minimal knowledge of actual plant component failure rates and test/maintenance intervals. As a result, many situations exist in operating plants where the availability of critical safety equipment is reduced due to excessive testing, which needlessly wears the equipment and increases its unavailability due to test/maintenance downtime. In other cases, availability is reduced because equipment failures remain undetected for long periods of time due to infrequent testing.

NU has a program which utilizes "best estimate" failure rates and actual plant statistics on test/maintenance downtimes to determine more optimum test and surveillance intervals. While this effort has been limited, the result of all systems studied to date has lead to valuable insights which have been incorporated into plant procedures.

A majority of the backfit projects at NU are in direct response to new NRC Regulations or changing interpretations of existing regulatory requirements. In some cases these requirements are found to conflict with each other or lead to hardware modifications which improve one aspect of plant safety while degrading another. To resolve these issues PRA techniques are employed by NU to:

(1) Assure net safety improvements result from proposed plant modifications.

(2) Assure optimum cost vs safety benefits are achieved when plant modifications are made.

(3) Provide justification for not pursuing those changes which have overall insignificant or negative safety impact.

G. Steam Generator Reliability

NU currently conducts extensive steam generator (SG) inspection and repair programs during refueling outages at its two operating PWR plants.

Multifrequency eddy current techniques are used to identify and quantify flaws in SG tubes, measure the height of sludge on the tube sheet, and measure the frequency and average size of SG tube dents. During the 1983 Millstone Unit No. 2 refueling outage, there were 2,139 tubes identified as containing flaws which exceeded the MP-2 tube plugging limit of "40 percent through-wall". These tubes have been either sleeved or plugged as appropriate.

Profilometer inspection techniques are used in analyzing the dents in SG tubes. The profilometer inspection of the SG tubes and strains. By monitoring the progression of denting between outages:

(1) The condition of the dents in the SG tubes is defined.

(2) Predictions of future tube dent sizes and strains can be made.

(3) The effectiveness of programs to arrest the tube denting mechanism can be evaluated.

Conservative SG tube-plugging criteria are used by NU to minimize the probability of a tube developing a through-wall flaw, which could result in a primary-to-secondary leak. The SG tube-plugging criteria are based on the dimensions of the tubing. Tubes must be either plugged or repaired if they contain flaws which exceed a specific percentage through-wall, restrict the passage of a specific size probe, or exceed a specific strain at a dent location.

To identify the corrosion mechanisms present in the SG's, tubes have been removed and subjected to extensive laboratory examination. Results of the examination are used to develop methods for eliminating further tube corrosion. For example, after defining the key role of copper in the pitting mechanism, sludge-lancing and chemical-cleaning programs were implemented. In addition, condenser air inleakage controls were introduced to minimize the ingress of feedwater copper.

In 1985 NU did an extensive chemical cleaning of the secondary side of each MP-2 steam generator up to the first support plate above the tube sheet. Because eddy-current testing was accomplished, both before the cleaning and after the cleaning, a cross-comparison of the effects could be accomplished. It basically showed that an additional 3,000 tubes needed to be sleeved due to "copper-masking" of pits in the tube material.

Optical inspection of the annular and tube lane regions is used to identify foreign objects which may be present. Upon identification, the location of the object is noted and an attempt at retrieval is initiated.

SG reliability is improved by programs such as these. They aid in identifying potential problems so that corrective actions can be initiated and safe, reliable operation of the SG can be achieved.

H. Review of Industry Experience

Industry experience is currently documented by several sources. It contains a wealth of material and is readily obtainable by utilities for review and evaluation.

1. Review of Failures at Other Plants

NU routinely reviews various nuclear utility operating experience information with the intent to help assure the safety and reliability of its nuclear units. Information originating from outside the NU organization generally takes the form of operating events or component failure reports. These reports are systematically reviewed and, when failures or events are found to be applicable and significant to NU nuclear units, cost-effective measures are taken to remedy deficiencies.

The following sources of utility experience information are reviewed:

(1) Technical Bulletins from NSSS suppliers. Typically, this information takes the form of Service Information Letters and Technical Information Letters from General Electric, Technical Bulletins from Westinghouse, and Availability Data Program Letters or Information Bulletins from Combustion Engineering.

(2) NRC Inspection and Enforcement (I&E) Bulletins, Circulars, and Information Notices. In addition, items identified as defective in the I&E correspondence are added to a NU Nuclear Operations Defective Items List (NODIL). Timely revisions to the NODIL are forwarded to the NU Quality Assurance Branch for use in minimizing the future procurement of potentially defective components.

(3) INPO's SEE-IN Program products, Significant Event Reports (SERs) that are transmitted to utilities over the Nuclear Notepad System and Significant Operating Experience Reports (SOERs) that are sent directly to the Senior Vice president of NE&O.

(4) NRC correspondence, such as Generic Letters, Abnormal Occurrences, Power Reactor Events, Inc.

(5) Other information on activities reported in INPO's Nuclear Notepad system.

(6) Industry and regulatory correspondence and reports, including Nuclear Operations and Maintenance Information Services (NOMIS), Nuclear Power Experiences, EPRI, and Nuclear Safety Analysis Center (NSAC).

(7) In addition, any occurrence at NU's nuclear plants, whether reportable or not, is recorded on a Plant Information Report (PIR). PIRs are reviewed at each plant for significance and reportability to NRC and are further reviewed in the corporate office by the Nuclear Plant Operations Analysis staff for precursors to the review data mentioned before.

Several examples of the effectiveness of these reviews come to mind. During May, 1980 INPO issued Significant Event Report 36-81 to utilities through the Nuclear Notepad system. SER 36-81 conveyed a fire hazard concern relative to no-load or light-load operation of the emergency diesel generators at Calvert Cliffs Unit No. 1. After reviewing SER 36-81 and finding it to be significant and applicable to all NU nuclear plants, an internal safety report was issued in September 1981. It identified the specific concerns and recommended procedural changes as corrective action. Responding to these recommendations, operating procedures were changed at NU's nuclear plants eliminating the fire hazard concern of SER 36-81.

During December, 1982 the NRC issued I&E Information Notice 82-50 to inform all nuclear plant licensees of a potential for misapplication of Brown-Boveri undervoltage relays Type ITE-27, Series 211B and 211L. Based on this information, NU reviewed the use of relays at our nuclear plants. We found that the reported relays either did not exist or were not misapplied at our nuclear plants. Further, as a preventative action, the relays were listed in the NODIL system. Listing in the NODIL system will required a review of the concerns expressed in I&E Information Notice 82-50 prior to any purchase of this type and series relay for nuclear plant application.

2. Statistical Comparison of Industry Availability

Availability data for the U.S. nuclear industry is obtained from several sources in order to compare the performance of NU's nuclear units with the rest of the industry.

NRC "grey book" data is utilized to compare the performance of our units with industry data in general, and with "sister units" more specifically. it identifies any deficient unit in our system that has a high potential for improvement.

NU maintains specific detailed outage files on all our systems and components. Statistical analysis of the data in terms of outage frequencies and repair times is performed and compared with industry statistics provided by the Westinghouse, General Electric, and Combustion Engineering data bases. The comparisons assist in promptly identifying which systems/components/parts perform below average. Once identified, corrective actions are recommended based on potential savings and costs.

The above type of analysis has resulted in significant savings to NU. For example, comparing the C-E statistical data on reactor coolant pump (RCP) seals with MP-2's RCP seal performance indicated, statistically, that the "B" RCP seals were very near the end-of-life. Although the seals had not been exhibiting any degradation characteristics, a recommendation was made to evaluate the seal with consideration of seal replacement during the next refueling outage. Close scrutiny of seal performance data (flows and temperatures) during subsequent operation did indicate degradation and seal replacement was scheduled.

I. NU Program Responsibility

The responsibility for the reliability activities described in Sections VI.A, B, C, D, E and H.2 are under Peter M. Austin, Manager of Reliability Engineering. Section F activities are under John H. Bickel, Supervisor of PRA. Section G activities are under Joseph M. Fackelmann, Supervisor of Nuclear Materials and Chemistry and Section H.1 activities are under Paul Callaghan, Manager of Nuclear Safety Engineering. These individuals also provided the written material used in the above descriptions. Further information in these subject areas can be obtained by contacting the cognizant manager/supervisor.

VII. Utility Industry Supported Programs

The utility industry has established and supported several organizations which serve the utilities on an industry-wide basis. EPRI was the first of these and works to coordinate industry research efforts. INPO and NSAC were formed following the TMI accident for the purpose of evaluating plant operating experiences, providing benchmarks for excellence in operations, and performing reviews of the adequacy of plant operation.

EPRI was established by the utility industry to conduct and sponsor research and development with respect to electricity production, transmission, distribution and utilization. The programs underway at EPRI cover a spectrum of topics that contribute to improvements in plant availability, reliability and safety. They include the screening of probabilistic risk assessment methods, advancement of nondestructive examinations and development of computer codes for monitoring reactor and plant systems. Utilities not only support these programs financially but also direct the research and development work through utility membership on the EPRI advisory committees. Some utilities take part in the actual performance of the work by supplying expertise and facilities.

The most significant of the utility industry initiatives are those carried out under the auspices of the Institute of Nuclear Power Operations (INPO). INPO was founded in the shadow of the TMI accident to raise the standards for nuclear plant operation. The founders delineated ambitious, but certainly achievable, benchmarks for excellence at commercial nuclear power plants. Over the past few years, INPO has been developing programs directed at implementing these goals. These programs include plant operation evaluations, the SEE-IN Program, Notepad, nuclear plant construction audits and training program accreditation. Clearly it can be stated that INPO both specifically and in a general sense performs as a safety and reliabilty conscience for the nuclear industry.

INPO has made great strides in the evaluation of nuclear power plant operations. Evaluation teams have traveled to nuclear plant sites throughout the U.S. Currently the teams are performing the fourth round of plant reviews. The INPO evaluation teams focus on major areas of organization and administration, training, operations, maintenance, radiation protection, chemistry and on-site technical support. In each area examined, actual conditions are compared to criteria for overall excellence, not to minimum acceptable standards. Management responses to-date have been positive and changes are currently being made in areas identified as needing improvement.

The INPO Significant Event Evaluation and Information Network (SEE-IN) program provides a comprehensive industry-wide assessment of operational events. The program was formulated in response to a recommendation of the Kemeny Commission, which stated: "There must be a systematic gathering, review and analysis of operating experience at all nuclear power plants coupled with an industry-wide international communications network to facilitate the speedy flow of this information to affected parties". Through the operation of SEE-IN, INPO evaluates nuclear plant operating experience by reviewing Licensee Event Peports (LERs). Recommendations for corrective action are disseminated to appropriate individuals and organizations on a worldwide basis.

The Notepad system, which is ancillary to the implementation of SEE-IN, provides an information inter-utility communication channel which transmits messages several times a day. It encourages the sharing of a broad spectrum of concepts, information and events. The system has ushered in a new era of openness and communication within the nuclear industry.

Another recent industry initiative instituted through INPO concerns the evaluation of nuclear plant construction projects. Industry leaders requested that INPO develop performance objectives and criteria along with a plan to evaluate the control of engineering, design and construction of plants less than 80% complete.

INPO also manages the Nuclear PLant Reliability Data Systems (NPRDS) providing an industry-wide data base for system and component operational history. They are also involved in a variety of other programs which have been established to enhance the reliability and safety of nuclear power plants.

Today, in addition to its member utilities, INPO has four nuclear steam supply system vendors and nine major nuclear engineering/construction firms as participants. They make important contributions to the events analysis review and assist in identifying precursors to serious problems. International participants now include 13 countries: Sweden, Germany, Belgium, Great Britain, France, Italy, Spain, Canada, Mexico, Brazil, Korea, Taiwan and Japan. The interdependence of nuclear facilities transcends national boundaries. The international agreements call for an exchange of information which will be of increasing value as more new developments originate overseas.

Further nuclear industry initiatives for reliable operation include the American Nuclear Insurers and Nuclear Electric Insurance Liability insurance pool. This program provides an extra incentive for utilities to adhere to INPO's safety standards. It offers monetary compensation against replacement fuel costs incurred by accident related shutdown of those nuclear units that conform to INPO's standards. In conjunction with the insurance program, the insurer performs evaluations and audits of the individual utility operations to assess their compliance to the safety standards and the exposure to financial risk.

The utility industry has frequently pooled their resources in addressing safety and reliability issues of common concern. Best known of these are the currently active "owners' groups". Though "owners' groups" have been in existence for some time, they have come into prominence since the TMI accident. Their popularity is due to their ability to find generic solutions to problems encountered by a number of utilities. The owners' groups have been established in many ways. Some have been organized through the nuclear steam supply vendor such as the General Electric, Combustion Engineering and Westinghouse Owners' Groups. Others have been constituted to address a specific technical problem such as the Steam Generator Owners' Group and the Industry Degraded Core Rulemaking Program. The owners' groups allow the utilities to draw upon their combined expertise or to contract programs whose solutions are beyond their individual technical capability.

VIII. Concerns with NRC Backfits

A major concern of NU has been the possible negative impact on the reliability of its nuclear plants caused by the numerous modifications mandated by the NRC.

During each of the first four years since the TMI accident, NU spent in excess of 100 million dollars for plant modifications. The majority of these modifications were NRC mandated projects related to post-TMI requirements and the Systematic Evaluation Program (SEP) upgrade work. NU has two nuclear plants which were subject to the NRC SEP.

I do not have data for the U.S. nuclear industry, but the NRC has indicated about 70% of the backfits are NRC required. I think this is a lower limit and know that most of NU's backfits are NRC mandated. Regardless of the exact percentage, the result is a continuous procession of plant modifications which were imposed rather arbitrarily without cost/benefit analysis nor agreed upon safety goals. I believe the changes have had an adverse effect on plant reliability and seriously question if they have enhanced safety.

It now appears that NRC management has recognized some of the mandated changes may not have improved safety. Thus, we are seeing a more disciplined approach by the NRC to the imposition of plant modifications.

This is manifest in the formation of the NRC Committee to Review Generic Requirements (CRGR). The CRGR has already rejected many proposed requirements that would have unnecessarily burdened utilities. More recently we have witnessed the adoption of draft numerical safety goals with which we can test the value of future NRC mandates.

The positive rationale provided by safety goals can be seen by the review of a recent NRC/Sandia report on MP-1 (SAND 82-2429). The report compared the resolution of SEP issues with the reduction in core melt frequency as calculated by the NRC in the MP-1 Interim Reliability Evaluation Program (IREP). The NRC/Sandia report addresses the resolution of 20 SEP issues and finds that 12 had no effect on reducing core melt frequency, seven reduced core melt frequency only 1-2% and one reduced core melt frequency by 16%. The report concluded "...Because of uncertainties in the data used in the Millstone 1 PRA, none of the effects (resolution of SEP issued) is at all significant compared to the overall uncertainty in the plant core melt frequency, exposure and risk..."

Clearly, the NRC's own studies show that in this instance there was negligible safety benefit associated with significant plant modifications. Unfortunately much of the modification work has been completed. However, for the future I am hopeful that these studies will provide a critical screening process, absent in the past, which will remove or modify those requirements which are not truly needed to achieve the appropriate level of safe and reliable operation of the plant.

To further enhance the understanding of backfits on the entire plant operation spectrum, NU has cooperatively developed the Integrated Safety Assessment Program (ISAP) with the NRC. This pilot program will look at all backfits, both those mandated and those initiated by the utility for reliability purposes. Utilizing such factors as a completed PRA of the plant, safety benefits, safety goals, systems interactions, and cost-benefit analyses, a ranking of all proposed modifications will be obtained by relative safety-significance improvements. It is hoped that this will allow both the utility and NRC to concentrate their efforts on those projects returning the largest safety benefit.

Millstone One is participating in the program in fiscal 1985 and will complete the study this fall. Connecticut Yankee is completing its PRA this year and will participate in fiscal 1986. It is hoped that the results of this pilot will be of great benefit to the industry as well as the NRC.

IX. Future Nuclear Plant Reliability Concerns

In Table I we saw the average capacity factor for NU's nuclear plants over the past eight years was about 72%. A pertinent question is what will it be in the future; will it increase, decrease or hold steady. NU has many ongoing programs, as I have discussed, which are directed towards detecting and correcting problems before they become severe. In addition, NU has purchased plant-specific simulators for each of its nuclear units to train operators and improve their performance. A further direct impact on capacity factor is being achieved by the extension of the MP-1 refueling cycle to twenty-one months. These programs are being supported because it is strongly believed they will enhance the reliability of NU's power plants.

In addition, a downward trend in the number of NRC mandated backfits that followed uncontrolled in the wake of the TMI accident is seen. I can only hope that the extent of the backfits to-date have not been overly harmful to the reliability of all plants.

Searching for a trend in capacity factors for NU's plants over the past eight years, I find none. The good performers remain good and the mediocre remain mediocre. So there is no apparent upward trend. Any improvement in that direction will have to come from current and future programs. Past efforts apparently have not been fruitful.

On the negative side, I see several potential problems which can only adversely affect reliability and capacity factor. The NRC's unrelenting insistence on a "fix" for the mythical ATWS event over the past decade is appearing nearer fruition. This will incur extended outages with no significant payoff in reliability nor safety. Also, those of us with PWRs, particularly those using sea water for condenser cooling and not having 100% leak-tight condensers, face a high probability of having to replace the tubes and tubesheets of steam generators sometime in the next decade. Another serious problem gradually descending on PWRs is the effect of fast neutron fluence on the nil ductility transition temperature (NDTT) of the beltline material in the reactor pressure vessel. This increased brittleness is gradually raising the minimum temperature at which we can pressurize the vessel. It has already impacted startup maneuvers and safety injection procedures at some plants.

Thus, looking at what the future holds, I believe we can achieve a small but significant improvement in nuclear plant reliability or capacity factors. NU is currently close to 20% above the world average capacity factor; the goal is to bring this figure closer to 25%, other factors remaining the same.

X. New Entrants into Nuclear Power

If I had the luxury of being a newcomer to commercial nuclear power today; i.e., the decision was made to enter the field, but no decision had been made as to organization, staffing or vendor, I would go relatively slowly.

First, I would visit several known good performers worldwide and particularly look at their organization to support their nuclear operations. Next, I would develop the skeleton of my organizational chart and fill the key management slots with known experts in nuclear power and management.

The next step would be a review of the problems associated with the plant classes of those organizations who are going to bid on the Nuclear Steam Supply System specification. Within their bid proposal, I would ask that they supply detailed information how each problem encountered has been solved for their NSSS.

This effort would take in the neighborhood of two-three years. Of course, in parallel the normal processes of estimating, specification preparation, approval, etc. would be ongoing.

Once the appropriate contracts, approvals, and permits are in place and construction has begun, the entirety of the nuclear organization must be totally staffed. It is especially important that the operating staff be selected early, since it is true that only they can do the "human-factors" review of their plant(s).

During the construction and pre-operational phases of the plant, a continuous review of worldwide problems must be accomplished. Only in this way can potential problems with your plant be ascertained and corrected, hopefully, before operation with consequent loss of reliability.

Once in operation, programs such as I have previously discussed must be rigorously carried out both to gain experience as well as improve upon reliability.

Obviously, reliability and safety go hand-in-hand; however, to have truly a safe operating organization, one must go quite a bit further.

XI. Safety Ethic

Safety is not written into the Code of Federal Regulations; it is only partially specified. It is at best only minimum requirements that, if met, should provide reasonable assurance that the public health and safety will be protected. So, how does one truly assure that organizations operating and maintaining nuclear power plants are doing so safely?

One way is to infuse the organization with a safety ethic. My definition of a safety ethic is that it is a state of mind which affects the entirety of an organization. It is a sense of responsibility and very strong professional attitude. Each person in the organization realizes that he is an important part of the big picture. Personnel within that organization question what may go wrong.

Within such organization each person feels a sense of responsibility to ensure each step of every activity is performed in a first class, professional and quality manner. People should understand that any mistake can be very costly, both from a financial point of view as well as from a personal credibility point of view. Such companies train their personnel well, ensure a knowledgeable, participating management and depend upon their people to do the job right the first time.

Organizations with a strong safety ethics expect their people to pay close attention to details. Operations personnel are alert and continuously question what might possibly go wrong. Engineers, when making design modifications, ask "what if the component breaks" questions; i.e., fail the piece of equipment in service and look at the consequence of the failure.

Studies have shown that 80-90% of the people interviewed after accidents of various types say they "didn't think" or "didn't realize" the consequences of their action. Good organizations with very strong safety ethics take the time to do it right the first time. The excuse "I didn't know" cannot be tolerated. Essentially, many of the findings from INPO and NRC involving personnel error are due to lack of attention to details or improper implementation.

When entering the nuclear power field, develop and implement comprehensive procedures for all aspects of the operation. Insist on an uncompromising commitment to follow directions and procedures. One of the largest quality assurance problems I have found in my experience is people not following procedures. Strong managements insist that people follow procedures or

- if the procedure is wrong, stop, have it changed and then restart the work
- if improvement is possible, follow procedure and then have it changed.

Another essential ingredient in maintaining a corporation in a position of leadership in safety requires a total commitment of the entire organization to safe designs and intense design reviews; to putting safety first--no shortcuts, no deviations from the first class way. It requires a system of many checks and balances. It requires diligent investigation of all accidents and near misses. People do not easily adapt to this atmosphere. They require the strongest kind of leadership from the top. This essential ingredient is called teamwork, which will assure that even the smallest detail has been addressed.

Each employee should know that he is an integral factor in the pursuit of excellence. With that, there can be no "pride of authorship". Checks and balances are definitely needed and, as I have indicated before, learning from the mistakes of others.

When developing your safety ethic, remember:

1. Regulations are minimum requirements
2. Insist on good procedures and train to them
3. Insist your personnel follow your procedures, and
4. Develop teamwork throughout the organization.

XII. Supervisory Responsibilities

Obviously, supervisors must assign their employees work, allocate other resources to meet corporate commitments, and provide early warning when commitments cannot be met. Sometimes it is forgotten that supervisors are not only management, but also leaders and trainers of their people. Supervisors must assure the job is done correctly and accept responsibility for what goes wrong. I would submit that a supervisor in a nuclear plant cannot do this while sitting in his office. He must be in the plant much of the day, leading and training his people.

In order to provide supervisors the opportunity to be in the plant, the administrative burden of paperwork must be removed from these positions as much as possible. Design changes to the plant must be done through a responsibile engineering organization. The plant staff should not be burdened with this responsibility except in a review mode. Similarly, the safety evaluations of such design changes should only be reviewed by the plant staff.

If we, as management, accept our responsibilities and set the tone for our nuclear plants to be operated and maintained to the highest standards of excellence, then I would submit that we can expect our personnel to follow that lead.

IV NUCLEAR UTILITY MANAGEMENT

Robert P. McDonald
Senior Vice President
Alabama Power Company

NUCLEAR UTILITY MANAGEMENT

Robert P. McDonald
Senior Vice President
Alabama Power Company

I live in Birmingham, Alabama, which is about 200 miles from our Farley nuclear plant. It's a wide road, it's not heavily traveled, and it takes about four hours of steady driving. So, as I was driving down the roads of Alabama for several hours, I had time to think about what it is that I have to give you that might be of some benefit. So what I'm going to give you is very personal. It speaks to principles which I believe in, which I try to use. And I say "try" because managing nuclear power plants is a very humbling experience. There is no place for arrogance. Arrogance is fatal in nuclear power. Managing nuclear power plants for high performance seems to me to be like an unending search for the truth. The truth about people, about institutions, about complex nuclear installations, and to me the truth about myself as a manager. I learn something every day.

So let me start and talk about what I consider the major principles. There are eight of them. These principles can be derived from the experiences of most successful nuclear operating people that I know. Although each utility has its own unique organizational structure as well as a unique location, within a given socioeconomic and cultural environment, these principles are believed to have a very high degree of commonality. I believe that each of these would stand up in Mexico, in China, in Portugal, in Egypt, in Brazil. At the same time, I know that if another senior vice-president nuclear power walked in here, he probably wouldn't agree with me on any one of them. So take them for what they are.

The first involves management's task of defining that which constitutes adequate performance. What do you really want to do? How well do you want to do it? What does "well" mean? What is management's commitment to do that? In the U.S., since Three Mile Island, adequate performance has been identified in terms of excellence. Excellence in English dictionaries means "by comparison". It means that, if I do excellent, there are two comparisons. One is comparison with what I've done before, and another is in comparison with what others do. It's a comparative term. So when we talk about excellence, when we talk about standards for excellence, we're really talking about a standard or an approach that is always trying to get higher and achieve better results than have been done before, on two fronts -- better than I've done or better than any of us all have done among us. At my company we use two means to define excellence and what our intent is with regard to performance. We call these the measures of excellence. I have passed them out for you to pick up a copy. It's a single sheet. The areas are: "margins of safety, effectiveness of public health assurance programs, reliability of power generation, cost of generated power". Those are the principles. We have on here: maximum time without an unplanned scram or trip; maximum time without any releases; maximum time without any radiation overexposures; maximum availability. In all those things that we seek to excell. And we have a list of the best that the Farley Nuclear Plant has ever done on them. We change it. The more we change it, the better we like it, because that means we are always improving something. That's how we define ours. We also define ours in terms of comparison with other plants.

INPO has just started putting out a set of curves, comparing one set of plant parameters with another. In the meantime, we, as with most plants, get a set of performance curves (there's about 30 involved altogether) to track the various parameters that support and contribute to this simple little bottom-line product.

Who uses these? These are posted on the door just inside the plant as you go in. The chief executive officer has it. Who uses these reports? The plant has them. They have all these posted on the board. Every manager has one, I have one, my chief executive officer has one. Everybody knows what we are trying to do. And we are committed. I like to say we have an invested commitment. If we say that we are going to be the best, we're going to do better, and we fail, then I lose some of my

JOSEPH M. FARLEY NUCLEAR PLANT
MEASURES OF EXCELLENCE

6-11-85

	HIGHEST ACHIEVEMENTS
A. MARGIN OF SAFETY FOR HEALTH AND WELFARE OF THE PUBLIC	
1. MAXIMUM TIME OF CONTINUOUS OPERABILITY OF NUCLEAR SAFETY SYSTEMS DESIGNED TO PREVENT OR MITIGATE SERIOUS SAFETY EVENTS AS NEEDED FOR EXISTING PLANT CONDITIONS. (PLANT BASIS)	CONTINUOUS FROM 10-28-82
2. MAXIMUM TIME WITHOUT AN UNPLANNED REACTOR TRIP OR SAFETY INJECTION. (PLANT BASIS)	321 ~~275~~ DAYS ~~1-14-83 - 10-16-83~~
B. EFFECTIVENESS OF PUBLIC AND EMPLOYEE HEALTH ASSURANCE PROGRAMS	
1. MAXIMUM TIME WITHOUT A RADIOACTIVE RELEASE TO ENVIRONMENT IN EXCESS OF TECHNICAL SPECIFICATIONS. (PLANT BASIS)	CONTINUOUS FROM 3-11-81*
2. MAXIMUM TIME WITHOUT A LOST-TIME INJURY. (PLANT BASIS FOR APCo EMPLOYEES)	174 DAYS 2-24-83 - 8-17-83
3. MAXIMUM TIME WITHOUT A PERSONNEL RADIATION EXPOSURE IN EXCESS OF FEDERAL REGULATORY LIMITS. (PLANT BASIS)	CONTINUOUS FROM 3-11-81*
C. RELIABILITY OF POWER GENERATION	
1. MAXIMUM AVAILABILITY (UNIT AND POWER CYCLE BASIS)	99.9% UNIT 2, CYCLE II
2. MAXIMUM CAPACITY FACTOR (UNIT AND POWER CYCLE BASIS)	97.3% UNIT 1, CYCLE VI
D. COST OF GENERATED POWER	
1. MINIMUM COST OF PLANT CONTROLLED PORTION OF POWER GENERATION EXPENSE. (PLANT BASIS. INDEXED FOR INFLATION & COST OF FUEL)	1.025¢/KWH 1982

SINGULAR ACHIEVEMENTS

E. SHORTEST REFUELING OUTAGE	37 DAYS, 22 HOURS UNIT 2, CYCLE II
F. LONGEST ON-LINE UNIT SERVICE	321 DAYS UNIT 1, CYCLE VI
G. LOWEST UNIT HEAT RATE (UNIT AND CYCLE BASIS)	10,866 BTU/KWH UNIT 2, CYCLE III

* NOTE: REFERENCE DATE FOR DATA IS FUEL LOAD ON UNIT 2 (3-11-81)

professional reputation. So I've got an investment in this business of trying to improve performance.

We have some other things. We have a signed contract with Duke Power Company that establishes a contest over about a year's period on who can have the best chemistry. We have a system to see who comes up with the best chemistry and, as you know, chemistry is very important in the operation of a nuclear power plant. We have comparisons with other companies which I'll talk to you about in just a little while.

Another of the comparisons which we have made has been with the French. We sent our best planners over to France about a month ago to review their outage plans. Outages are very interesting events. This picture which I am unfolding and will give the video people exercise with (laughter) ... Tom, we're out of space. Tom, you're going to have to back out. Now this is a master plan, a partial plan, of the critical path for a refueling outage. This is only a little part of it. Each one of these blocks has a supporting network that shows how it's done. So we took a plan like this to France and compared it with theirs, because we want to have compact, high-quality, short outages. That's just one of the ways, an example, of how we look at objectives, how we look at what we're trying to do in the way of improving performance.

The second principle is the establishment and maintenance of an environment that fosters the accomplishment of the desired performance. Such an environment normally is one which includes all parameters, not just availability, not just cost, but everything you set out to do, because if you don't try to get everything done you set out to do, you and your people will sense that you are not really sincere, that you are willing to sacrifice one for the other. So when you think about what you're trying to do, you have to think about it carefully and not put in there little things that don't make much difference. This environment thing should be one where management is essentially obsessed with quality and high performance. I warned you earlier that I was going to speak personally, but I feel strongly about these things, because when you talk to your people, the vital link with your performance is your obsession with doing the job right and doing it better. That's the commonality that must be between you. This idea of an environment, as I thought about this group, was very intriguing, because there's going to be some differences among us. First I thought of the plant environment. I think we would all agree that a plant should be clean. It should be orderly. There should be good evidence that things are maintained well. It should also be arranged for people to work. If you can't have people work, they're going to say, "What the heck, let it run." So you've got to have it look like you expect high performance. In the nuclear organization, which Mr. Burstein talked to you about, if the public relations man has more authority than the plant manager, you've got problems.

There's another type of problem. If you have an organization with one nuclear unit in it, beware! There you've got a problem. With two nuclear units, you've got a problem. You know, it's like trying to raise children. People frequently have more problems raising one child than they do a half a dozen, because they seem to support each other. There tends to be peer pressure, so that when one makes a mistake the other one does not make it. When one does something right and is rewarded, the other one does it. They learn from each other. So I think that's a very strong characteristic. We in the U.S. are 55 different utilities; we had to have an INPO to tie us together. We had to have it. And so when you create a nuclear organization, you should try to have as many units as you can and you should try to have them as standardized as you can, so that you have lots of things in common. I cannot stress that enough. When you're one small utility, you must plan for and expect to have a more difficult time managing that one unit than you would probably have in managing two. You may have more total problems, but they won't be total new problems. They won't be problems which you can't handle.

The next item on environment has to do with the company environment. If you introduce a nuclear organization in a utility company, and you interface that with other functions of the company, fossil, hydro, gas, etc., those parts of the company will probably not have the same value system and set of working principles that the nuclear has. They will probably not have the same career paths, or skills values. And so, when you're working with people in an environment where they are in essence some outsiders inside your company, you can find some problems. You need in some way to use every effort possible to manage those interfaces so that the nuclear organization sees solid company support and dedication to good nuclear principles of management. Very important. It's gone wrong in a lot of U.S. companies.

Then there's the last subject, and that is the cultural environment. You may not be very aware, but there's quite a difference in the cultural environments within various places in the United States. The Yankees and the Rebels. Man, there's all the difference between a plant in southern Alabama and one up in Illinois or the Northeast. On the one hand, it's the good old boys from down South who take things kind of easy, and the Yankees from up North who are hard-headed and won't listen to anybody. So there's a lot of differences in our nations, and I'm sure there will be in yours. There is also the question of whether you use indigenous or imported people. Most utilities that I know try to use indigenous people. By that, I mean people who are from that area. If we get a Northern boy in our plant, young, training to bring him up, the chances of his staying there are probably only 50%. That means increased turnover. That means I have to hire more people. It loads the training program down. It increases my average expense. Also, the guy from up North, he doesn't get along quite so well. He doesn't communicate his desire in the same words. He doesn't like to talk slow like down South. He'll say, "Why don't you go ahead and spit it out?" And somebody gets mad at him. We are blessed with a common language. Some people wouldn't call it a common language, but it's supposed to all be English. Where you might end up with various dialects or languages you can have tremendous problems.

There's another problem which I have run into in some countries which I'll call a "class factor". In some countries there is a reluctance for supervision or management to ever be caught with their hands dirty. If you have that problem, you have a real handicap, because, I'm going to get to another factor down here near the end which says that the managers and supervisors in this business have to be very technically competent. A manager simply cannot comprehend what it means when a valve was not opened properly unless he's been there. I frequently tell the story, when I was in the Navy -- I went to the U.S. Naval Academy -- and I had a roommate, and when we were first year people, we were hazed. And one of the things we had to do was to open the radiator for heat each morning during the winter. And some of the valves were upside down, some were on the side, and some were on top. I've been hazed many times because one of my roommates couldn't tell which way to turn the valve. And he went four years. He never could understand how to turn a valve the right way when it was upside down. When I was a young officer, I was aboard a diesel-powered submarine and was coming out of Yokusko harbor, getting ready to submerge. We rigged the submarine for "dive". We had a large valve in the overhead. I had to check many valves, and that was one of them. I know I checked that valve in the open position. When we went to submerge, the submarine took a large tip. We came back and that valve, which had a chain and lock on it, was in the wrong position. I saw another one. I was in a submarine going around the world submerged one time, a new nuclear submarine. We were in the middle of the Indian Ocean. We had a plant problem and we were going to cross-connect one generator with another. When you synchronize to get the phases in, you wait until the arrow comes around and throw the switch. And I stood there over my electrician, and I saw him, and he synchronized 180 degrees out of phase. Everything went black. We had no power in the middle of the Indian Ocean. 300 feet down. Unless you have some experience, unless you know how mistakes are made, unless you know how to approach the business of avoiding them, you are not going to make it in the management of nuclear power. So, if you're working in a society which has a class distinction about whether you get your hands dirty or not, you've got a problem. You may be able to overcome it, but you sure ought to be aware of it.

Well, let's get on to the next principle. That involves what I like to call management by problems. Therein, management is continually searching with a type of instinctive skepticism about the apparent well-being of things. If you don't have that skepticism, you walk through a plant, and it's shiny, its running -- Gee, everything is great today. Little do you know that right around the corner is the start of a disaster waiting to happen. If you're not looking for it, it'll find you first. So this idea of managing by problems is to be instinctively skeptical, to go out looking for them, to identify them, figure out what to do with them, and then resolve them. Then, after you do that, you have to keep that history alive. You need to make an environment so that people learn by their experience, so it's just handed down from there on. This involves an attitude and an approach that is characterized by respect and, again, humility, for the magnitude of the problem of running a complex nuclear plant. It's a continuous admission that we have imperfect management; that we have to work every day at problems to be able to reach high performance. When I was in diesel submarines in our Navy, people would tell you how good they were. They never told you problems. If you said, "Well, I went out on exercises," they'd say, "How did it go?" "Oh, it was great, no problems, no problems." Now if you ask a nuclear manager how it's going, he'll say, "Oh, maybe better or worse. Here are my problems."

We periodically put out this "Major Problem Sttus Report". It tells everybody a lot of mistakes I've made. It tells what we're doing. it goes out to everybody, even my chief executive officer. That's the way it's got to be. Has anybody here ever read a little book or an article called, "I'm OK, You're OK"? Well, it's a unique little book about 15 years ago, and it talks about how to feel good about yourself. And, in order to feel good about yourself as a nuclear manager, you better find some problems to talk about, because that's the only way you'll ever feel comfortable.

The fourth principle involves extensive, comprehensive, and continuous plans. Such planning is necessary for high performance. It has to be done in near-term and long-term integrated schedules -- there's simply no way to go out and wing it and get high performance. By "winging it" I mean, playing cowboy, hot-rodding it. I hope one of those words strikes a chord with you. But it means doing the best you can without any forethought. Don't do it. You have to have detailed planning and daily work schedules. You just can't go and say, "I'm going to work on a valve." You've got to arrange for it to be isolated. You've got to arrange for the pressure to be taken off. Arrange for contamination control. Arrange for proper inspection. You've got dozens of things you can't just go do. That's a lot of work to integrate that into a program.

Refueling outages. I've showed you this. There's a lot more than that to it. Let me tell you a little story about this one. We got serious about four years ago about outage plans. Typical outages had been running very long. So I said we were going to cut down. So we started trying to cut this down. This represents about a 38-day outage. We thought we could get it down at that time, so I finally worked with people who came up with the concept of optimum scheduling. That doesn't sound very revolutionary. But what it means is that, when you figure all these little activities on that critical path, you don't put in there what you think you'd like the time for the job to be. You put in there the best demonstrated past performance of that job. Not the worst, not any extra, but the best. The optimum that it's ever been done. If you'd like to talk about that some time, I'll talk to you about what it does for you. It puts people on notice, and you get better quality than you've ever had before.

Training -- planning and training. Manpower. Preventive maintenance. And perhaps the one thing which is somewhat maligned, people think of nuclear power people as living by the book. Military people, living by the book. Two different kinds of books. Nuclear power people must live by the book, because the book is usually sets of procedures. In our plant we have over 3000 procedures, but each procedure represents in written form a planned out way to get something done. It was planned by experts in a time of cool reflection, in a room or a lab, thought out in detail, organized into steps. It was a plan. And by using that plan, call it procedure, you're chances of getting the job done right and right the first time are much higher than if you didn't have anything like that. So one of the principles is, if you want to err, err on the side of planning.

The fifth principle is that of persistent training. I'm not going to talk much about training, but let me say this. If you're going to commit one-sixth of your people to training, if you recognize the importance of training, it should be approached with the same type of deliberate actions that every piece of work is done. That in essence, I think, is what I described in performance-based training. I think we, in the U.S. industry, are doing the job right for the first time. You know, it's not a new industry in the U.S. I've been in nuclear power for twenty-nine and one-half years. But never have I done training like what I'm doing today. It must be done that way. It can't be short-cut. Persistent training.

The sixth principle is one which I believe in very strongly, and that is the use of the fundamental principle that in all organizational structures when you assign work you assign two things -- you assign it with responsibility and accountability. And you assign it to one person. Responsibility and accountability to one person. That's sounds so simplistic. It sounds like a freshman management professor. But if you look at any organizational design, you'll find it's not so. If you find that you do do that, people cooperate laterally very well. And they cooperate vertically. Let me tell you one area that I've observed where it is misused time after time in the nuclear industry, and that's in quality assurance. One of QA'sthe roles is to conduct audits; that is an independent group, sits on the side, performs no work, and it audits performances. When they audit, they write out a report. Findings. Here's what's wrong. And then in some cases they tell people what to do. Now they're not in a position, most probably, to be as highly skilled and trained in that activity as the person doing it. But they're telling him what to do. Then they send that report, not to the guy who's responsible, but about five levels above. I know one company where all those reports go to the chief executive officer. And they go up to the guy who's responsible for financial matters. And then, when the plant takes it, they write out their correction report. Who's it go to? It doesn't go to their boss. It goes to the QA guy. That QA guy is not accountable for how that work's done. His job is to do auditing. That's all. And if you hold that QA guy accountable for that plant running properly, there's no way he can do it. You've got a full-time guy running it. Think about it. Apply the rule of responsibility and accountability throughout the organization, and you have a lot tighter-run process. It helps you put your finger on who's to blame and whom to reward. If you assign a job to a bunch of people, another favorite target is you have a high-level nuclear review group. That's all right if the review group is an advisory to an official. But if you have to send stuff back and forth to it, you're looking for trouble. We know none of those review group members is accountable. He'd say, "Hey, I'm the chemistry specialist. There isn't much chemistry in that." Another guy says, "Why, yeah, I said you should do that, but I hold a job over here in another department." It's just crazy. This is a good principle. I highly recommend it.

The seventh principle is the need to have adequate engineering, technical, and material support. Typically, any given utility cannot directly employ enough people with required engineering and technical knowledge to safely and efficiently maintain their nuclear power plant. Not to operate, but maintain it. Also, it's not cost-effective to do so. It's not cost-effective to maintain enough spare parts and material not to have to go outside and get parts once in awhile. You'd have a whole new plant, sitting one beside the other, and you'd use the second plant for spare parts. You need ready accessibility, the external engineering and technical expertise. I hold this to be one which is the least sophisticated in the U.S. In France, where you have a large number of plants, they have an arrangement where they have a service company; it works very well. I think Japan and some other countries do very well. In the U.S., we're very versatile, with a lot of expertise, but they're servicing a lot of those 55 different utilities. I believe that a given utility needs to select a very few number of service organizations and build a relationship with them like a joint business venture to where there is some continuity, not responsibility for running the plant, but responsibility for providing the services, and it's a very firm business arrangement. I look at some utilities where, during a refueling outage, they have 20 or 30 contractors on site. No way can you manage that many contractors. No way! We have found it helpful to form business team arrangements with our major vendors, and have a round-the-year type relationship where we do planning for outages, planning for problems, and keep up a support relationship. I suggest to those of you who don't have a broad base that should be one of the things that you're very careful about, very careful about, because the day's going to come as it came last night at my plant, and I didn't hear about it until this morning. What happened? We were steaming along at 100% power, and one of the throttle valves started to go closed. And the power would dip down. About five minutes, again. The supervisor got on the phone to an engineer, saying, "Look, I got a problem. I got a valve that's going 'Ssshhh'." He said, "Well, the response time is so and so; we could take the replacement card. You can probably pull it out and jerk the other one in and catch it and everything will be all right." That happened. Perfectly safe. The diagnosis for that problem in that period of time was simply not available in our plant. You need some more depth. I tell another little story. About two weeks ago, we had an intermittent control-rod drive power problem, where, when we pulled the control rods, when they supposedly got to the top, there was a group of them that went all the way up there. And we couldn't find them. We put electronic monitors on it, we did every kind of test, we couldn't find it, because it was intermittent. Sometimes it would be there, sometimes it wouldn't. It would be on another .....? So we started from Saturday noon and had a dialogue with four engineers. We continued testing. At 8:00 Sunday we decided, "We don't know what we're doing. We're not getting anyplace." So we chartered a plane and we flew in the technical team on site. There was a woman in that plane. And when we got down to a crucial decision, we had one of 16 paths to go. Each path took about six hours. It could have taken 96 hours to get that problem. And she says, "I think you ought to try that one. I saw something like this happen in another plant one time, and I think it's that." So we tried it, and that was it. She saved us at least four days of operation at peak power. That's a lot of money. That's up in the millions of dollars. You can't afford <u>not</u> to have that type of support.

Now let's get to the last principle. The last principle I think is the most important. It's the establishment and nurturing of a management team that possesses the unique capabilities needed for adequacy in nuclear power plant management. Each manager involved, without exception, must have adequate knowledge, qualification, and experience in nuclear power as a speciality area. He must be able to participate energetically with effectiveness and knowledge and involvement in finding the problems, avoiding the problems, and managing the enterprise. He must maintain a high discipline in his organization. He must give it a strong sense of direction. He must be able to connect with people on what you're trying to do at any one time. He must have the confidence of his people, so that there isn't anyone who is ever afraid to speak out. Everyone feels like they're required. If something's wrong that they don't know about, they're required, it's expected of them to speak up about it. And you can only do that when you have a knowledgeable manager. You know how many people you have worked for who didn't know the area you were working in, and you were reluctant to talk about the problems because you knew he didn't understand. It happens all the time. It has particular importance for utilities in the start-up phase. This has to be from top to bottom. There's no room for loitering at the top, because the problems can be squashed by the people who don't understand the problem. Just look at what has happened in the U.S. We start off with nuclear plants, and they're treated like fossil plants. And then they decided, "Well, we better separate them a little bit." So they did assign another department under the nuclear plant. Then, as time went on, they said, "No, they need their own general manager." So they split them apart there. Then they said, "No, they need their own executive." So they split them. "They need their own higher executive." So more and more you see a full recognition that they are different. It appears to me that the U.S. is moving closer and closer to specialized nuclear operating companies. And I think that is really the only answer in the long run.

In conclusion, it's a tough challenge. All these principles. I don't think I talked about anything that any of you would find new. I don't think I told you a new thing. I hope I put it in perspective for you. I hope I talked a little bit about how they fit together. I think the business of managing these plants is vital, vital to all of us. The importance of energy, the cost of energy in the world. And we are our brothers' keepers. We have a vital interest in the affairs of each other.

V TRAINING OF NUCLEAR FACILITY PERSONNEL

Forrest J. Remick
Professor of Nuclear Engineering
Associate Vice President for Research
The Pennsylvania State University

TRAINING OF NUCLEAR FACILITY PERSONNEL

Dr. Forrest J. Remick
Professor of Nuclear Engineering
Associate Vice President for Research
The Pennsylvania State University

It's an honor and a pleasure to participate in this conference. My first experience in teaching goes back around 29 years, in the school that was called the International School of Nuclear Science and Engineering, conducted at Penn State University and North Carolina State University in conjunction with the Argonne National Laboratory. This was a follow-up of President Eisenhower's Atoms for Peace speech in the United Nations. The program was started to bring people from foreign countries to the United States and give them an intensive program in nuclear engineering. I taught for several years in that program, teaching reactor physics and reactor laboratory, and I know that at least one of you in the audience attended that program in its later years.

I plan to talk a little bit about the legal and regulatory background for qualification and training in the United States, and then discuss some recent industry initiatives in the area of training and some recent industry experience.

You probably know that the Atomic Energy Commission in the United States was established in 1946 by the Atomic Energy Act of 1946, but in this country when we refer to the Atomic Energy Act we're really talking about a major modification that took place in 1954, at the time of Eisenhower's Atoms for Peace talk after which there was major declassification of the information and a decision that this would be put out into industry and universities and shared with foreign countries, and so forth. So when we talk about the Atomic Energy Act, we mean the act of 1954 as amended. That particular act gave the Atomic Energy Commission, which had been established in 1946, the authority to prescribe uniform conditions for licensing individuals as operators, determine the qualification of such individuals, issue licenses to such individuals, and suspend such licenses for any violations.

TRAINING OF NUCLEAR FACILITY PERSONNEL

Introductory Comments

REGULATORY BACKGROUND

Legal Authority

- AEC established by AEA of 1946

- Atomic Energy Act of 1954 (AEA), as amended, authorized the U. S. Atomic Energy Commission (AEC) to: (Sec. 107 AEA)

  - prescribe uniform conditions for licensing individuals as operators

  - determine the qualifications of such individuals

  - issue licenses to such individuals

  - suspend such licenses for violations

- Energy Reorganization Act of 1974 (ERA) separated the AEC into the Energy Research and Development Administration (ERDA) and the U. S. Nuclear Regulatory Commission (NRC).

  - Effective January 1975

  - ERDA later transformed into U. S. Department of Energy (DOE)

  - ERA Section 202 transferred the licensing authority from the AEC to the NRC

Rules and Regulations of the NRC

- Contained in Title 10, Chapter 1, Code of Federal Regulations - Energy (10 CFR)

- In particular, two parts are applicable to training

  - 10 CFR Part 50 (10 CFR 50) - Domestic Licensing of Production and Utilization Facilities (in part)

The AEC operated for about 28 years, quite successfully from my standpoint. We came to a period of time, in the early 1970s, when there was a movement in this country of some disenchantment with nuclear energy, and there was a vocal minority of people who looked at the Atomic Energy Commission and saw that that one agency of government had the responsibility for the development of nuclear energy, the promotional aspect, and also the dual authority to regulate. And this gave some people some concern. I do believe it was a minority, and so forth, but in 1974, Congress passed what was called the Energy Reorganization Act of 1974, which separated the AEC into two new agencies. One, the Energy Research and Development Administration, which we called ERDA, was to handle the developmental side, including the weapons side, of the old AEC. The other was the U.S. Nuclear Regulatory Commission, established to handle the regulatory or licensing functions. This went into effect in January, 1975. Since that time, a little over ten years, we have the Nuclear Regulatory Commission, which has the regulatory authority for administering licenses and so forth for nuclear energy in this country. ERDA, the Energy Research and Development Administration, did not last long. There was immediate disenchantment with ERDA, and it was transformed within a couple of years into the U.S. Department of Energy, DOE. The Energy Reorganization Act, in Section 202, transferred the licensing authority that had rested with the AEC to the NRC. It's my personal view that the demise of the AEC and particularly the fact that there was one joint Congressional committee that had oversight responsibility over the Atomic Energy Commission, the demise of that structure has had major impacts on nuclear energy in the United States. I believe the Nuclear Regulatory Commission has about seven or nine oversight committees of Congress that give advice on a day-to-day basis.

The actual rules and regulations of the Nuclear Regulatory Commission, which implement the laws as outlined in the Atomic Energy Act and the Energy Reorganization Act, are contained in Title 10, Chapter 1, of the Code of Federal Regulations, the one marked "Energy". We refer to that normally as 10 CFR. There are two parts of those regulations that apply to training. One of them only in part, and that is Part 50 of 10 CFR, which is entitled "Domestic Licensing of Production and Utilization Facilities." A part of that refers to training and qualifications of personnel.

risk?" Because, in order to justify further change, what should I use? What benefit is to be obtained for the cost of further reducing public health risk when public health risk is so low? By any measure I cannot justify any change. I'll tell you one change I might be able to justify. I might be able to justify a little lower on core melt if you let me count the billion dollars worth of damage to him, not to the public -- the economic damage to the owner. If I can consider that, I might be able to justify some greater reliability, but is that my job? That's not safety regulations. Our objective is assure safety, and, as I've said before, once I reach the level where public safety is assured, then let the economic forces determine the appropriate efficiency. I have minimum regulation for safety. Our requirements must be stable and predictable. We went through a great difficulty in the United States in the 1970s and the early 1980s because we developed nuclear regulation in the 1970s and had ever increasing requirements, and then, when TMI happened, we had a massive increase in the number of requirements. And the owners were never sure of what are the requirements. They're unstable. You cannot plan. You cannot predict. And they would start to build something, and halfway through building it, they might be told "That won't be acceptable; you'll have to go back and start over." That's where cost comes in. That instability is very bad. So we seek stable and predictable regulations based on risk, where we are really providing protection to the public and not wasting public resources. Remember, every dollar that's spent on an unnecessary improvement of a nuclear power plant is not a cost to a company. It's a cost to the electricity company, the company's customers, the people. The people pay for it. It's just like a tax. Someone has to pay for that, and if it an unnecessary, a frivolous thing, it serves no good. It is an unnecessary thing, and it's just wasted money. We are not doing anyone any good with that. So we do not want to have unnecessary requirements. And, of course, we now have this wealth of experience, so that we can have both risk-based and experience-based standards. You're going up to Idaho, some of you, and you will hear that the research information we have beneath and supporting our regulations is enormous now. So much better than even ten years ago. So we have this desire and this ability to have

- - 10 CFR Part 55 (10 CFR 55) - Operators' Licenses (in its entirety)

- In addition, guidance is provided in a variety of other NRC documents, including Regulatory Guides, NUREG documents, etc.

- Only two categories of personnel are licensed by the NRC (Slide 1)

  - - Reactor Operator (RO) - any individual who manipulates a control of a facility (i.e. apparatus or mechanisms which directly affect the reactivity or power level of a reactor) (10 CFR 55.4 (d) & (f))*

  *Citations from the Rules and Regulations are paraphrased for simplicity and hopefully for ease of understanding.

  - - Senior Operator (SO) (more frequently referred to as Senior Reactor Operator (SRO) - any individual who directs the licensed activities of licensed reactor operators (10 CFR 55.4 (e)).

- <u>10 CFR 50 Requirements</u> (for power reactors) (SLIDE 2)

  - - No one except an RO or an SRO may manipulate the controls (10 CFR 50.54 (i) and 10 CFR 55.3 (a & b)).

  Exception - an individual who manipulates the controls as part of his/her training to qualify for a license under the direction and presence of an RO or an SRO. (10 CFR 55.9)

- - Within 3 months after issuance of Operating License (OL), licensee must have in effect an operator requalification program, which as a minimum meets the requirements of 10 CFR 55, Appendix A (10 CFR 50.54 (i-1)).

- - Note - oddly, there is no stated requirement to have in effect a training program for personnel seeking a license, although it is understood and all nuclear utilities have established formal training programs.

(SLIDE 3)

- - An RO or SRO must be present at the controls at all times that the facility is in operation (10 CFR 50.54 (k)).

on only those that have to do with power reactors. There are some interesting requirements that I'd like to highlight. For example, it says that no one except an RO or an SRO may manipulate the controls. That certainly makes sense. If you're going to require a license, only licensed people should be able to manipulate those controls. There is one exception, however, which applies to power reactors, and that's any individual who manipulates the controls as part of his or her training to qualify for a license under the direction and presence of a licensed RO or SRO. There is one other exception which you might be interested in, which applies to universities, where students in a formal nuclear engineering course can also manipulate the controls of a research reactor as part of their training. Also, Part 50 says that within three months after a utility receives its operating license the licensee must have in effect an operator requalification program which, as a minimum, meets the requirements of 10CFR Part 55, Appendix A. Appendix A is entitled "Requalification Programs" and outlines one of the requirements in requalification. By requalification I mean retraining. I'm going to say more about that requalification training program later. It's interesting to note, at least from an academic standpoint, that, although in the regulations there is a specific requirement that utilities or licensees have a requalification program, there's no similar requirements stated that you must have a training program. This is kind of a quirk; somewhere along the line it got overlooked. That does not mean that people do not have training programs; every utility has a training program. Every licensed reactor in the country, whether power or non-power, has training programs, but there's not a requirement. The only thing that is said is that in your Preliminary Safety Analysis Report you must describe your training programs, and so forth, but it doesn't require it. The important point is that everybody does have a training program.

Continuing on the Part 50 requirements, an RO or an SRO must be present at the controls at all times the facility is in operation. That certainly makes sense. Operation is usually defined in Appendix A of the actual license, which we call "Tech Specs". In Tech Specs they'll tell you when that facility is considered in operation and when it is you must meet that requirement.

The next requirement is that a minimum of two ROs and two SROs per shift must be on site whenever a nuclear power unit is operating. That is a fairly recent requirement; in fact it went into effect in January, 1984. Now I have actually simplified this, because the number of SROs and the number of ROs that are required is dependent on the number of units at the site, the number of those units which are operational, the number of control rooms, and whether they're common or not. So it's much more complex than I have here. For example, if you have two units that are operating and you have one control room for those two operating units, then you need two SROs and three ROs. If you have two units operating and there are two separate control rooms, then you must have three SROs and four ROs. And it goes on in that way.

There's also a requirement that an SRO must be present in the control room at all times that a unit is in operation. That is a recent addition, which I believe went into effect also in January, 1984. That was a particular requirement which I personally tried to get softened a bit, because of the requirement "at all times". In other words, if there's an SRO in the control room and he needs to go to the restroom, he cannot do it unless another SRO relieves him. And I personally feel that that's a bit strict but the Commissioners themselves decided that this was going to be a requirement. So now, if somebody wishes to leave the control room, there must be somebody to replace him.

A Shift Technical Advisor (STA) is also a requirement which was added following the accident at Three Mile Island. That person is an advisor to the shift supervisor. He's required to be on site. The STA <u>must</u> have a bachelor's degree, it says "or equivalent", but I do not think any "or equivalent" has been approved. So basically it's being implemented as "must have a bachelor's degree in an engineering or scientific discipline". By the way, the clause, "or scientific discipline" is the reason why Mr. Counsil said that you could have a geologist or geoscientist as an STA, and it doesn't make sense. In addition to a degree in an engineering or scientific discipline, the STA must have received specific training in the response and analysis of the plant for transients and accidents. After the Three Mile Island Unit 2 accident, when people looked over what the

•• A minimum of two ROs and two SROs per shift must be on-site whenever a nuclear power unit is operating. (The staffing for multi-unit sites is dependent on the number of units, the number operating and whether there are common control rooms). (10 CFR 50.54 (m-2i & m-2ii) including table.)

•• An SRO must be present in the control room at all times that a unit is in operation. (10 CFR 50.54 m-2iii)

(SLIDE 4)

•• A Shift Technical Advisor (STA) to the shift supervisor is required on-site The STA must have a bachelor's degree or equivalent in an engineering or scientific discipline and have received specific training in the response and analysis of the plant for transients and accidents. (NUREG - 0737, Item 1.A.1.1)

NOTE - this is not a rule or regulation, but was voluntarily complied with and inacted by confirmatory orders of the NRC.

•• It is possible that licensees will be permitted an alternative to substitute STA with SRO for the second SRO on shift. (Under discussion by NRC Commissioners).

• 10 CFR 55 Requirements for Operators' Licenses. (SLIDE 5)

•• Medical examination is required to assure that physical condition and health of applicant are not such as might cause operational errors. (10 CFR 55.11 (a)).

•• An applicant must pass written examination and operating test (or simulated operating test) administered by the NRC (10 CFR 55.11 (b))

•• License is limited to the facility for which it is issued (10 CFR 55.31 (b)).

(SLIDE 6)

•• License expires two years from date of issuance (10 CFR 55.32)

•• License will be renewed if individual (10 CFR 55.33 (c))

- passes new medical examination.
- has satisfactorily completed requalification program
- has discharged his license responsibilities competently and safely.
- is needed as an RO or SRO at the facility.

operating crew had been doing (and by the way, I think you'll find that that was probably an above-average competent crew that was on shift at that particular time), each one of the operating people was very busy doing things at the console, making changes, and so forth, in a fairly competent manner. However, there was nobody who stepped back and looked at the broad picture of what was happening and tried to determine what event was actually taking place. So the NRC came up with the idea that you should have somebody who's an advisor to the shift supervisor who has broad engineering background, and who will step back and look at the broad picture, won't get involved with what's going on minute to minute. This is how the Shift Technical Advisor came about. It's controversial. You heard what Mr. Counsil said. He does not believe that it's a very good idea to have an STA. There are mixed views on that. I would guess about 50% of the people say that it makes sense to have a stepped-back Shift Technical Advisor, who's not involved in operation, and 50% who feel that, no, it's better to have that person as part of your operating crew. I tend to agree with Bill Counsil on this, at least up to a point. The NRC does not require that this person be licensed as an SRO. The problem can be that you have a young engineer come in who has a very good college background, by the way, and he's going to be in the background and tell these people who have been running that plant, say, for 10 or 15 years, what to do, or advise the shift supervisor what to do. In many cases, this person is not readily accepted and is not credible to many of the operators. I personally think that, if you're going to have STAs, they should be a part of the crew and be licensed as SROs. If they've gone through the training programs, if they must be requalified as regulations require, and so forth, the chances are that they are going to be more accepted. It's interesting that, although every utility must have a Shift Technical Advisor, there is no formal regulatory requirement in Part 50 or 55 that they have one. It came about after Three Mile Island. You'll find it in a NUREG document, which the NRC just sent out under "confirmatory orders". So they all have it, but it's not a regulatory requirement. There's a very good chance that the Commission will issue a regulation which will incorporate the STA but give the utilities an option. Either have a separate STA, as they now have, with no requirement for licensing, or if they wish, to have this person licensed as an SRO. Then they can use that person as the second SRO that's required to be on shift. In the last year or so that this has been discussed, I've had the opportunity to talk to a number of utilities, and once again they come out about 50-50, where some of them say, "Even if we're given the option, we will keep a separate STA because we think it's a good idea." By the way, some of those who do have STAs do require them to get licensed. And the others say, "No, when that option comes along we will take this person, he will be our second SRO, and then, procedurally, in cases of a transient or an accident, that second SRO will have the responsibility of stepping back and looking at the broad picture." So this is a possible change.

If we then look at 10CFR Part 55 (Slide 5) and some of the requirements in there that might be of interest to you, there is a medical examination required to assure that the physical condition and health of the applicant are not such as might cause operational errors. That certainly makes sense. An applicant must pass a written examination and an operating test, or a simulated operating test, administered by the Nuclear Regulatory Commission. Not all countries require reactor operators to be licensed. Japan is one of those that does not require reactor operators to be licensed. I'll say more about that examination a bit later. Further, the license is limited to the facility for which it's issued, so if you look at the plant that Mr. Counsil came from, Millstone Units 1, 2, and 3, those were different types of units, so if you got a license there, you would have a license for Millstone 1, a license for Millstone 2, or a license for Millstone 3. Now it's possible to have 1 and 3, but you had to qualify in each one of those. You could not get one license and say that you could use it in all three plants. You had to actually acquire separate licenses. Sometimes there are waivers if you go from essentially an identical plant to a sister plant.

The license expires two years from date of issuance. That means a license is good for two years. Now, I would expect that some time within the next four to six months the NRC will issue a change to its regulations making that license good from somewhere between four and six years. As it is now, the license will be renewed at the end of that two-year period if that individual passes a new medical examination, which means you must take a medical exam every two years, has satisfactorily completed the requalification program that I mentioned earlier, has discharged his license responsibilities competently and safely, and that the facility states that he is still needed as an RO or an SRO at that particular facility.

Let's look then at some of the eligibility requirements to be a reactor operator in the United States. They're in the most up-to-date form and most concise form in NUREG 1021. The experience requirements are that you must have a minimum of two years of power plant experience, of which at least one year shall be nuclear power experience, and you must have a minimum of six months at the site for which the license is sought. So that's basically the minimum experience requirements to be a reactor operator.

If we look at the training requirements, you must spend a minimum of three months' training in the control room. The next slide shows some of the things that were added post TMI-2. You must receive training in heat transfer, in fluid flow, in thermodynamics, in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged, and in reactor and plant transients. I think you can see, if you know something about the Three Mile Island accident, why it's necessary that people get training in those. Also, following the requirements in Part 55, the NRC expects that people who are going to sit for reactor operator licensing examinations have had a total of 500 hours of lectures on principles of reactor operation, design features, general operating characteristics, instrumentation and controls systems, safety and emergency systems, standard and emergency operating procedures, radiation control and safety procedures. In addition, one must have completed an NRC-approved training program of at least one week's duration at a nuclear power plant simulator. One must have manipulated the controls of the facility for five significant reactivity changes and participated in reactor and plant operation at power levels of at least 20%. The minimum educational requirements for reactor operators is a high-school diploma or equivalent.

(SLIDE 7)

ELIGIBILITY REQUIREMENTS - REACTOR OPERATOR
(NUREG - 1021, ES - 109)

- Experience Requirements

  - Minimum of two years of power plant experience of which at least 1 year shall be nuclear power experience.

  - Minimum of 6 months at site for which the license is sought.

- Training Requirements

  - Minimum of 3 months' training in the control room.

(SLIDE 8)

  - Training in:
    - heat transfer
    - fluid flow
    - themodynamics
    - use of installed plant systems to control or mitigate an accident in which the core is severely damaged.
    - reactor and plant transients

(SLIDE 9)

  - Total of 500 hours of lectures on:
    - principles of reactor operation
    - design features
    - general operating characteristics
    - instrumentation and control systems
    - safety and emergency systems
    - standard and emergency operating procedures
    - radiation control and safety procedures.

We now look at the requirements for a Senior Reactor Operator, once again coming from the NUREG document. The experience requirements depend on whether you have a college degree or you do not. It's stated here, "assuming a college degree" and then in parentheses I've added the requirement "if you do not have a college degree." If you do have a college degree, a minimum of two years of responsible nuclear power plant experience, which may be as a staff engineer involved in the day-to-day operation of the plant. If you don't have a college degree then it's four years of experience. So you get credit for two years of that experience if you have a college degree. You must have a minimum of six months at the site for which the license is sought if you have a college degree and one year as a licensed reactor operator if you do not have a degree. The training requirements, in either case, are a minimum of three months on shift in training for an SRO position. You must also, following the Three Mile Island experience, receive training in heat transfer, fluid flow, thermodynamics, use of installed plant systems to control or mitigate an accident in which the core is severely damaged, training in reactor and plant transients, reactor theory, handling and disposal of radioactive materials, specific operating characteristics of the plant, fuel handling and core parameters, and administrative procedures, conditions, and limitations. It's also expected that you will have 500 hours in formal lectures in some of the same subjects: principles of reactor operation and reactor theory, design features and specific operating characteristics, instrumentation and control systems, safety and emergency systems, standard and emergency operating procedures, administrative procedures, conditions, and limitations, radiation control and safety procedures, handling and disposal of radioactive materials. And similar to a reactor operator, you must have satisfactorily completed an NRC training program of at least one week's duration at a nuclear power plant simulator, you must have manipulated the controls of the facility during five significant reactivity changes, and participated in reactor and plant operation at power levels of at least 20%.

The educational requirements are little unclear, but basically as stated in Slide 14. The minimum requirement is a high-school diploma, or

(SLIDE 10)

- - Satisfactory completion of NRC approved training program of at least one week duration at a nuclear power plant simulator.

- - Manipulation of the controls of the facility during five significant reactivity changes.

- - Participation in reactor and plant operation at power levels of at least 20% power operation.

- <u>Education Requirements</u>

  - High school diploma or equivalent

(SLIDE 11)

<u>ELIGIBILITY REQUIREMENTS - SENIOR REACTOR OPERATOR</u>
(NUREG - 1021, ES - 109)

- <u>Experience Requirements</u>

  - Minimum of 2 years of responsible nuclear power plant experience which may be as a staff engineer involved in the day-to-day operation of the plant. (4 years, if no degree)

  - Minimum of 6 months at the site for which the license is sought. (1 year as RO, if no degree)

- <u>Training Requirements</u>

  - Minimum of 3 months on shift in training for an SRO position

- Training in: (SLIDE 12)

  - heat transfer
  - fluid flow
  - thermodynamics
  - use of installed plant systems to control or mitigate an accident in which the core is severely damaged
  - reactor and plant transients
  - reactor theory
  - handling and disposal of radioactive materials
  - specific operating characteristics of the plant
  - fuel handling and core parameters
  - administrative procedures, conditions and limitations

(SLIDE 13)

- Total of 500 hours of lectures on:

  - principles of reactor operation and reactor theory
  - design features and specfic operating characteristics
  - instrumentation and control systems
  - safety and emergency systems
  - standard and emergency operating procedures
  - administrative procedures, conditions and limitations
  - radiation control and safety procedures
  - handling and disposal of radioactive materials.

(SLIDE 14)

- Satisfactory completion of a NRC approved training program of at least one week duration at a nuclear power plant simulator.

- Manipulation of the controls of the facility during five significant reactivity changes.

- Participation in reactor and plant operation at power levels up to at least 20% power operation.

• <u>Education Requirements</u>

A 4-year degree in engineering or applied science, or equivalent, or (a high school diploma, or equivalent).

equivalent, but of course some people have a four-year degree, and therefore the requirements are slightly different. However, as I stated, that educational requirement is not very clear.

Let's look a moment then at the examination that you must take if you wish to be a reactor operator. There's a formal, written examination. It used to be eight hours in length. The NRC now has shortened that to six hours. I'm not sure if the exam is shorter; I think people are just given less time. They certainly look as long as they were when the requirements allowed people eight hours. The examination covers four categories of questions, the principles of nuclear power plant operation, thermodynamics, heat transfer, and fluid flow. There is a section on plant design, including safety and emergency systems, sectional instruments and controls, and on procedures including normal, abnormal emergency and radiological control. The requirement is that you must receive at least an 80% score on the overall examination, and you may not have less than 70% in any one of those four categories. In fiscal year 1984, 77% of the people who took that examination after the training passed the examination. Now, this particular organization of the examination is fairly recent. If you look in the regulations themselves, they will list 12 categories but in the last couple years the NRC has taken those 12 categories and grouped them into the four I have just given you. So there is no real change there in requirements. It's just a grouping in the examination where each one of those sections counts for 25% of the examination.

In addition to the written examination, it's required that you take an operating and oral walk-around examination. If you have a simulator, chances are that you will take between a two- and four-hour demonstration examination at a simulator, and then a four- or five-hour examination at the plant, walking around and demonstrating to the examiner that you know the plant and what the various items of equipment are for. If you do not have a simulator, then it is all done at your own plant, and it would typically take about six hours. In fiscal year 1984, approximately 89% of the people who took the operating and oral walk-around examination passed it. A total of 639 people took that oral examination in 1984.

REACTOR OPERATOR LICENSE EXAMINATION

(SLIDE 15)

- Written examination
  - 6 hours in length
  - Covers following four categories*
    - Principles of Nuclear Power Plant Operation, Thermodymanics, Heat Transfer, and Fluid Flow
    - Plant Design, including Safety and Emergency Systems
    - Instruments and Controls
    - Procedures - Normal, Abnormal, Emergency, and Radiological Control
  - 77.1% passed in FY 1984

*10 CFR 55.21 lists 12 categories. Currently these are combined into the above four categories for implementation (NUREG - 1021, Rev. 1, Section ES-202)

- Operating and oral walk-around examination
  - 4-5 hours in length
  - 89.2% passed in FY 1984

SENIOR REACTOR OPERATOR LICENSE EXAMINATION

(SLIDE 16)

- Written examination
  - 6 hours in length
  - Covers following four categories*
    - Theory of Nuclear Power Plant Operation, Fluids and Thermodynamics
    - Plant Systems: Design, Control and Instrumentation
    - Procedures: Normal, Abnormal, Emergency and Radiological Control
    - Administrative Procedures, Conditions and Limitations

    * 10 CFR 55.22 lists 9 categories. Currently these are combined into the above four categories for implementation (NUREG - 1021, Rev. 1, Section ES-402).
  - 81.3% passed in FY 1984

If we look at the Senior Reactor Operator, a written examination is required, once again six hours in length, down from formerly eight. It covers four categories of questions: theory of nuclear power plant operation; fluids and thermodynamics; plant systems, design, control and instrumentation; procedures, including normal, abnormal, emergency, and radiological control; and administrative procedures, conditions, and limitations. Once again, if you look at the regulations you actually see nine categories listed there, but they've been grouped, once again, into four categories for implementation in recent years. And 81.2% of the people passed that examination in 1984. In 1984, the NRC administered 746 SRO examinations. So it's no small job just to examine operators in the United States. It's a large effort. Remember that the plants are distributed around the country, and the examiners have to go and administer these at the site. An operating and oral walk-around examination is also required for an SRO. That was not required before Three Mile Island, but since TMI, the operating and oral walk-around examination is required. It also is about four to six hours in length. In fiscal year 1984, approximately 92% of the people passed that portion of the examination. Over many years, it averages out to about 85% of reactor operators passing the examination and, luckily, 90 to 95% of senior reactor operators passing. Of course, in some utilities, it's disastrous -- few pass. In some utilities, 100% of them pass.

Let's look at the requalification program requirements. Appendix A to Part 55, says that you must conduct a requalification program on a continuous basis, with a period not to exceed two years. So the requalification program is required to be in place three months after you have an operating license for the plant. It must be designed to run continuously over no more than a two-year period and then be continued throughout the life of the plant. There's a requirement that it must include pre-planned lectures, so you cannot have a requalification program where somebody takes videotapes or self-study material and did it on his own. It must include pre-planned lectures. It may include individual study if you wish. There's no objection to that, but that cannot be 100% of the requalification program. It must include on-the-job training in such things as

- Operating and oral walk-around exmination
  - 4-6 hours in length.
  - 91.8% passed in FY 1984

REQUALIFICATION PROGRAM REQUIREMENTS (SLIDE 17)
(10 CFR 55, Appendix A)

- Conducted for a continuous period not to exceed two years.
- Must include preplanned lectures
- May include individual study, in part
- Includes on-the-job training (OJT)
  - Manipulate control
    - -at least 10 reactivity manipulations
    - -SRO at least direct manipulation by operator
  - Knowledge of equipment and procedures
  - Knowledge of facility design, license and procedures changes
  - Review abnormal and emergency procedures
  - Simulators may be used

(SLIDE 18)

- Evaluation
  - Annual written examination
  - Routine written exams on subjects studied
  - Observation of performance of duties
  - Simulation of abnormal or emergency conditions
    - Discussed at console
    - Demonstrated at simulator
- Accelerated retraining if individual shows shortcomings.

manipulating the controls of the facility. You're to have at least 10 reactivity manipulations during the two-year period, if that's the length of your requalification program. In the case of the Senior Reactor Operator, he does not necessarily have to manipulate the controls, if he is supervising a Reactor Operator in manipulating those controls. You must demonstrate a knowledge of the equipment and procedures, and a knowledge of any facility design, license and procedural changes that have taken place. You must review abnormal and emergency procedures on this two-year frequency. Simulators may not only be used; their use is encouraged.

Evaluation of the requalification program: it's required that there be an annual, written examination, patterned after the examination given to operators, when they originally sit for their license. In addition, routine written examinations are to be given on any subjects that they study during this two-year period. There's to be an observation of the performance of their license duties. There's also to be a simulation of abnormal or emergency conditions -- what they would do following these procedures. That could either be discussed at the console, if a simulator does not exist, or demonstrated on a simulator. For the annual examination the requirements are that you have at least 80% overall and no less than 70% in any category.

In recent years, the Nuclear Regulatory Commission has decided that they will no longer allow utilities to administer these annual examinations on their own. In the past, the utilities administered the examinations, and the NRC came around from time to time and audited. They would take out an examination, look it over, grade it themselves, see if the utility was grading too easily, and so forth. But if the questions were somewhat similar to what the NRC would ask, if the grading was somewhat consistent, they found that satisfactory. The last couple years, the NRC decided that it will go out and conduct a portion of those annual requalification written examinations itself. So they have a practice now where they will go out and give NRC examinations to 20% of the operators at 50% of the facilities each year. So each year, they are giving requalification examinations to 10% of the operators, both Reactor Operators and Senior Reactor Operators in the U.S. This has a dramatic effect on morale of operating personnel. They fear an NRC examiner coming and giving the examinations. Some of these operators have been operators for 10 or 15 years, and they say it's like an MD having to take his examinations every year, or a Ph.D. required to take his comprehensive examinations every year, and so forth. So there has been a morale problem. People also claim that not many of the NRC examiners have actual utility operating experience and, therefore, their questions tend to be more theoretical and not performance-based. So right now, there's considerable discussion about the whole concept of requalification. Nobody's saying that requalification training is not good, but the fact that the NRC is actually going and administering these has caused people to raise the question of what should be in a requalification examination. So it's possible that in the next couple years this will change somewhat.

I'd like to talk now a little bit about the training and accreditation initiatives that have taken place in the last couple years by the industry itself. Although the NRC has some requirements on the qualification and training of people, there's no question that the responsibility for conducting that training resides with the facility licensees themselves. No question about that. All facility licensees have training programs at least for licensed personnel, and by that I mean for Reactor Operators and Senior Reactor Operators. However, other than for ROs and SROs and more recently STAs, if training has existed for other personnel such as mechanical maintenance, electrical maintenance, instrumentation control, health physics technicians, chemistry technicians, and so forth, if it existed at all, it was much less formalized in the past.

Following the Three Mile Island accident, the industry formed the Institute of Nuclear Power Operation, INPO, in order to pursue and attain quality and excellence in nuclear power plant operation. As part of that activity, INPO pursues a large number of activities to assist its member utilities. I'd like to identify a few specific examples which relate to training activities of utilities. For one, INPO has developed 17 training and qualification guidelines, which are based on input from the industry and on a systematic analysis of jobs in key nuclear nuclear plant positions. These guidelines outline the course content needed for the

TRAINING AND ACCREDITATION

- Responsibility for training of nuclear facility personnel resides with the facility licensee.

- Although requalification programs are required by the NRC

    - Training programs not mandated (quirk of regulations)

    - However, all facility licensees have training programs

    - More formalized for licensed personnel and STAs than for others

- Following TMI-2 accident in 1979

    - Industry formed Institute of Nuclear Power Operation (INPO) in order to pursue and attain quality and pursue excellence in nuclear power plant operation

    - INPO pursues a number of activities to assist its members.

- The following are specific examples of INPO training activities:

    - INPO has developed 17 training and qualification guidelines based on input from the industry, and the results of a systematic analysis of jobs and tasks in key nuclear plant positions. These guidelines outline the course content needed for the training and qualification of personnel in the nuclear power plant positions. These guidelines are as follows: (SLIDE 19)

        - Pressurized Water Reactor Control Room Operator, Senior Control Room Operator, and Shift Supervisor Qualification

        - Boiling Water Reactor Control Room Operator, Senior Control Room Operator, and Shift Supervisor Qualification

        - Nuclear Power Plant Requalification Program for Licensed Personnel--Guidelines for Requalification Training and Education

        - Nuclear Power Plant Non-Licensed Operators -- Guidelines for Qualification Programs

        - Technical Instructor Training and Qualification training and qualification of personnel in the nuclear power plant positions. These 17 guidelines are shown on Slide 19. The 17 guidelines that they have issued are performance-based, that is, based on a close analysis of the jobs and tasks performed in the industry. The first of these is for pressurized water reactors, control-room operator, senior control-room operator, and shift supervisor. Control room operator here is just another name for Reactor Operator. Senior control room operator is just another name for Senior Reactor Operator. They have the same thing for boiling water reactors. Next is a guideline for a nuclear power plant requalification program for licensed personnel -- guidelines for requalification training and education. They have a guide for non-licensed operators, and for technical instructors, people who teach in the training programs. There's a simulator training guideline. For nuclear plant staff, there are guidelines for heat transfer, fluid flow, and thermodynamics instruction, what these people should know in those areas. There are guidelines for training to recognize and mitigate the consequences of core damage, a requirement following Three Mile Island. They have guidelines for the position of Shift Technical Advisor, recommendations for the position description, qualifications, education and training. They have technical development programs for technical staff and managers; guidelines for mechanical maintenance personnel, electrical maintenance personnel, instrumentation and control technicians, radiological protection technicians, chemistry technicians, general employee training, and guidelines for quality control inspectors and non-destructive examination technician training. These guidelines have all been issued in the last several years by INPO and they are very helpful, particularly to new utilities bringing plants on line.

INPO has also performed a detailed job analysis, followed by a detailed task analysis of key nuclear plant positions. Job analysis determines the tasks that are performed by each position and task analysis determines the knowledge and the skills needed for each position. The job and task analysis identifies training requirements for plant personnel and aids the utilities in making sure that their training programs are comprehensive or, we say, systematic. I don't know how many of you are familiar

- Simulator Training Guidelines
- Nuclear Power Plant Operating Staff -- Guidelines for Heat Transfer Fluid Flow, and Thermodynamics Instruction
- Nuclear Power Plant Operating Staff -- Guidelines for Training to Recognize and Mitigate the Consequences of Core Damage
- Nuclear Power Plant Shift Technical Advisor -- Recommendation for Position Description, Qualifications, Education, and Training
- Technical Development Programs for Technical Staff and Managers
- Guidelines for Mechanical Maintenance Personnel Qualification Programs
- Guidelines for Electrical Maintenance Personnel Qualification
- Guidelines for Instrument and Control Technician Qualification
- Radiological Protection Technician Qualification
- Chemistry Technician Qualification
- General Employee Training
- Guidelines for Quality Control Inspector and Nondestructive Examination Technician Training

•• Also, using industry expertise and experience, a detailed job analysis, followed by a detailed task analysis, has been conducted for key nuclear plant positions. Job analysis determines the tasks performed by each position and an analysis of these tasks determines the knowledge and skills needed for each position. The job and task analysis identifies training requirements for plant personnel and aids utilities in making sure that their training programs are comprehensive.

(SLIDE 20)

Job and task analysis has been completed for the following positions:

- shift supervisor
- control room operator
- senior control room operator
- plant equipment operator
- instrument and control technician
- chemistry technician
- radiological protection technician
- electrical maintenance technician
- mechanical maintenance technician

with job and task analysis. Basically, it's this. You take a position; it might be that of reactor operator, senior reactor operator, shift supervisor or health physics technician. You look at a large number of people who hold that position in the industry and find out from them, their supervisors, from the procedures they must follow, and so forth, what jobs they do have in their position. In other words, what things must they do as a reactor operator. And you make a list of all these jobs. Then you look at all the tasks to do a particular job. You turn on this switch, you turn that knob, you go inspect this, and so forth, and you list these. You do this with a large number of operators, and you come up with a list of all the tasks to perform the jobs and the position. And out of this you find that to perform the jobs of this position, this person must have certain knowledge and he must have certain skills. He must twist this knob, he must reach out here and do that, and so forth. So you come up with a list of knowledges and skills that this person must have to carry out the tasks and the jobs of this position. Sounds like it makes a lot of sense. Then you ask yourself where a person gets that knowledge, or where a person acquires that skill? Does he get that knowledge in a college-level course? Does he get it in a high-school chemistry course? Is the only way he's going to get it from a special course, lectures, and so forth? Where is he going to get the particular skills and where is he going to be trained to get those skills?

Then you take that information and you say, "I'm going to design a training program to make sure that that person has all that skill and all that knowledge." You do it by sitting down and writing learning objectives. This person must have this knowledge or he must have this skill. I'm going to write a learning objective, what it is that I need to train him to know or to do. Then you ask yourself, how am I going to know that he's now acquired this knowledge or this skill? So, along the way you set certain criteria that he's got to meet. And then you ask how am I going to examine him to know that he has that knowledge now, or he has that skill? So your examination goes back, once again, to what he has to do. We say it's performance-based. Then you design the training program, either using a lecture, using videotape, using on-the-job training, using self-study, or what, to accomplish that. And all along the way, you are giving him examinations to see if he has learned this information. Not only that, you must continuously evaluate the program by talking to the trainee to see if he has learned it, talking to his supervisor when he goes back to the workplace to see if he has learned it. If not, come back and provide that input to the training program. That is systematic, performance-based training. I apologize for taking so long but sometimes there are words whose meanings are not clear. So when we talk about job and task analysis or systematic performance-based training, that is what we are talking about, and that is what INPO has implemented in the U.S. utilities or is in the midst of implementing at the moment.

INPO has performed job and task analysis for a number of positions, and Slide 20 shows positions for which they have now completed job and task analysis. Tremendous undertaking! This information is all computerized and members of INPO, including foreign members, have access to this information on line of all these tasks, jobs, skills, knowledges, and so forth. These include shift supervisor, control-room operator, which is a reactor operator, senior reactor operator, plant equipment operator, which sometimes is called auxiliary operator or non-licensed operator, instrumentation control technician, chemistry technician, radiological protection technician, electrical maintenance technician, and mechanical maintenance technician. Many of these are done specifically for PWRs and also a different list for BWRs. In addition, the Institute sponsors a number of workshops and special seminars for utility training personnel to assist them in developing their own training systems. They conduct workshops on training and accreditation, and for chief executive officers, plant managers, training managers, and so forth. I personally have participated in training sessions for chief executive officers and for their plant managers. This November there's a workshop with the chief executive officers. It's quite an interesting experience to go and sit down in a workshop with the chief executive officers of all 55 U.S. nuclear utilities. It's a unique opportunity to make some points.

INPO also conducts what they call "evaluation and assistance visits" to utilities, where, on a frequency of about once every 15 months, they

- - The Institute sponsors workshops and special seminars for utility training personnel to assist them in developing their own training system. Training and accreditation is often a major subject at other workshops such as those for utility chief executive officers, nuclear plant managers and other technical disciplines.

- - During its regular plant evaluations for each utility, INPO evaluates the training programs and the work of plant personnel to ensure continuing training quality. INPO observes and evaluates the actual workplace application of the knowledge and skills taught in the classroom and in simulator training.

All of these activities are designed to assist member utilities as they improve and upgrade their training programs for nuclear plant personnel.

- After TMI, utilities launched agressive programs to upgrade the training for nuclear power plant personnel

- INPO and industry determined that an accreditation program was needed to evaluate and verify that these nuclear plant training programs were achieving high standards.

- INPO began developing objectives and criteria for accreditation, and in 1982 issued the first formal criteria for the Accreditation Program. A revised set of criteria was issued in 1985.

- Each of INPO's 55 member utilities -- every U. S. utility that owns, operates or is constructing a nuclear power plant -- has committed to achieving and maintaining accreditation, not only for licensed operators, but for all of the 10 key positions involved in nuclear power operations.

- These positions are as follows: (SLIDE 21)

  - nonlicensed operator
  - reactor operator
  - senior reactor operator/shift supervisor
  - shift technical advisor
  - instrument and control technician
  - electrical maintenance personnel
  - mechanical maintenance personnel
  - chemistry technician
  - radiological protection technician
  - technical staff and managers

send a large team, 12 to 15 people for two weeks, to the plant. These teams are made up of INPO staff members and also peer evaluators from other utilities with a Senior Reactor Operator's license. They go into the plants for two weeks and look very closely at the operations. They look at the training programs and the performance of operators and then they write a report to the utility. This is a very powerful technique. The first time that this was done, retired Admiral Dennis Wilkinson, who was the president of INPO, went to every one of those site visits, every single one of them, the first time around. It is unique to have one person visit all U.S. nuclear facilities in a short period of time and meet with the CEOs and tell them what they think about their plant. Very powerful! There's nobody in the NRC who has that knowledge and no one who can have such influence on the Chief Executive Officers as the President of INPO.

All of these activities are designed to assist the utilities. INPO is not a regulatory body. It's a creature of the industry itself, and therefore it is not in there to give them fines, to enforce regulations, and so forth. It's to assist them to seek excellence and improve quality.

After the Three Mile Island accident, the utilities launched an aggressive program to upgrade their training programs because one of the findings was that training programs were not adequate. INPO and the industry soon jointly determined that, if they were going to improve training programs, there had to be some way of accrediting training programs. There must be some objectives and criteria established and somebody must judge whether these training programs meet those objectives and criteria. So they said there should be some kind of accrediting program, just like there is for engineering programs at universities in this country. Universities have an organization which comes around; they look at your program and you are accredited or you're not accredited, in accordance with a set of objections and criteria. It can mean a lot to you if you are accredited. And it can mean a lot to you if you're <u>not</u> accredited because, generally, certain jobs for engineers are not available if you have not graduated from an accredited institution. If INPO training programs were not accredited, there could be implications from insurance companies, and so forth.

So INPO began developing objectives and criteria for accreditation. In 1982 they issued their first formal set. However, early in 1985, they issued a revised set. The current set is in a booklet called INPO-85-002. This provides the 12 objectives and the very many criteria under each objective of what it takes to have an INPO accreditation program.

Each of INPO's 55 member utilities, and that's every U.S. utility that owns or operates or is constructing a nuclear power plant, has committed to achieving and maintaining accreditation, not only for licensed reactor operators, but for all of the ten key positions involved in nuclear power plant operations. These positions (Slide 21) are the positions of non-licensed operator (these are the auxiliary operators, or the equivalent operators); reactor operator position, senior reactor operator/shift supervisor position; shift technical advisor; instrumentation control technician; electrical maintenance personnel; mechanical maintenance personnel; chemistry technician; radiological protection technician; and technical staff and managers.

To be accredited, the training programs must be based on a systematic approach and must be performance-based. That is an absolute requirement. This means that they must contain the following essential ingredients, shown on Slide 22. That is, they must be based on a systematic analysis of jobs to determine what tasks the performer of the job must be able to perform. Performance-based learning objectives, derived from that analysis, must be developed. The training must be designed, developed, and implemented to achieve those performance-based objectives. Evaluation of the trainees must be conducted during the time that they're being trained. Such evaluation should be made against the performance standards stated in the learning objectives. An evaluation of the effectiveness of the training program should be conducted. Such evalution should include provisions for revision of the training program based on the trainee's demonstrated ability to perform in the actual job setting. Those are the basic five ingredients of an accredited training program for those ten positions.

The accreditation process itself has three steps. First, using the accreditation objectives and criteria in the booklet that I just showed

(SLIDE 22)

- Training programs must be based on a systematic approach and must be performance based . In general, this means that they must contain the following essential ingredients:

  - A systematic analysis of jobs to determine what tasks the performer of the job must be able to perform.

  - Performance-based learning objectives derived from the analysis.

  - Training designed, developed and implemented to achieve the performance-based objectives

  - Evaluation of trainees conducted during training. Such evaluation should be made against performance standards stated in the learning objectives.

  - Evaluation of effectiveness of the training program. Such evaluation should include provisions for revision based on trainees' demonstrated ability to perform in actual job setting.

- The first utility training programs were accredited in August 1983.

(SLIDE 23)

- The accreditation process has three steps:

  - First, using the accreditation objectives and criteria, the utility performs a self-evaluation to identify and correct weaknesses in its training programs. The utility writes a comprehensive report, which is provided to INPO, that describes how the accreditation criteria are being met.

  - Second, an accreditation team, made up of training experts from INPO and other utilities, visits the plant and evaluates the training programs. The team's recommendations, along with the utility's responses, are presented to the National Nuclear Accrediting Board in a written report.

  - Third, the decision to award or defer accreditation is made by the National Nuclear Accrediting Board.

  - The National Nuclear Accrediting Board includes four categories of members:

    - senior utility representatives
    - non-nuclear training experts
    - representatives from the post-secondary educational community
    - individuals nominated by the U. S. Nuclear Regulatory Commission

you, the utility must perform a self-evaluation to identify and correct any weaknesses in its training programs. The utility then writes a comprehensive report, which is provided to INPO, that describes how the accreditation criteria are being met. To me, this is the most important part of the accreditation process. To do it correctly, you must have the training personnel at the utility and the plant operations personnel and some of the very best of your plant operations and training personnel sit down and take an objective look at what your training programs consist of and what they should consist of. And then you have to be honest with yourself in saying, "Are there weaknesses, and what can we do to correct them?" And you must undertake them. So a report is written. INPO then takes this report and puts together an evaluation team. Last week I sat in on a couple days of evaluation of a two-unit site in the Northeast. There were 15 people on the evaluation team, not counting myself as an observer, not counting a person from the NRC as an observer. These people were there for one week. They prepared themselves in advance. Chances are they have been to that utility several times providing them advice, helping them along the way, because their purpose is to provide assistance. However, they then go to the site for one week and make an independent check. They look at the report but they also look in depth at the training programs. They track down things like records, they track down the qualifications of instructors, whether those instructors have a professional development program, whether they have the minimum qualifications for the position. They look at all aspects. They come back and write a report, an evaluation report, which is sent to the utility. In that report, the INPO evaluation team will make recommendations, and by recommendations they mean things that do not meet the criteria or objectives of INPO accreditation. The utility has an opportunity to then respond to those comments.

That brings the third step into play, which involves the National Nuclear Accrediting Board. The accrediting board looks at the self-evaluation report from the utility. It looks at the evaluation report and the utility's responses. It meets with the representatives of the INPO evaluation team and, typically, three to six senior management members of the utility which is having its programs accredited. The National Nuclear

- The majority of the Board members are from outside the nuclear utility industry.

- Five members constitute a Board.

- The National Nuclear Accrediting Board is charged with seeing that nuclear plant training programs meet the INPO criteria.

- When training programs come before the Board, members examine the report of the accreditation team and the utility's responses, as well as the utility self-evaluation report.

- The Boards's final decision comes only after its meeting with representatives from senior utility management, including management representatives from the plant and training staffs.

- INPO requires a status report every two years after the utility achieves accreditation.

- Full reaccreditation is required once every four years.

- All 55 nuclear utilities with operating plants have committed to have their 610 training programs (10 programs at 61 nuclear power plant stations) ready for accreditation by the end of 1986.

- New plants will seek accreditation within two years of receiving their full power operating license.

- To date, 68 training programs at 16 plants have been accredited.

Accrediting Board includes four categories of members: it includes some senior utility representatives; it includes some non-nuclear training experts (for example, one person who's on the accrediting board is a former vice-president for training from United Airlines, a former 747 captain and an FAA-approved check pilot); it includes representatives from the post-secondary education community, in other words people from universities who have been involved in the accreditation of university engineering programs; and it includes individuals nominated by the U.S. NRC. I serve on the accrediting board in that latter category, somebody nominated by U.S. NRC. The board consists of five members; there's a pool of 18 people, from which five are selected. There are two utility people, one non-nuclear training, one post-secondary person, one NRC nominee, who make up a board. The majority of those members, therefore, are not nuclear utility representatives. The National Nuclear Accrediting Board is charged with seeing that the nuclear plant training programs meet the INPO criteria. They have the final decisional authority. When the programs come before the board, the board members examine the report from the accreditation team, the response, the self-evaluation report, and their final decision only comes after they meet with the representatives of the senior utility management, including representatives from the plant and training staffs.

The first utility training programs were accredited just two years ago this month, in August, 1983. That was the OCONEE plant of the Duke Power Company. After one is accredited, one is required to submit a status report every two years after the utility achieves the accreditation, so OCONEE's first report is due this month. And full re-accreditation is required every four years, so the utilities must go through this process every four years.

It's possible that other positions, other than the ten I mentioned, will be added to the accreditation program. It will not happen before the first go-around, but there is consideration of formal accredited training programs for quality assurance personnel.

All 55 nuclear utilities with operating plants have committed to have their 610 training programs (that's ten programs at 61 nuclear stations in this country) ready for accreditation by the end of 1986. That is a major commitment. New plants that came on line since December, 1984, have committed that they will seek accreditation within two years of the time they receive their operating license. To date 68 training programs, at 16 plant sites, have been accredited. A little of over 10% of the commitment. Only one plant has had all ten of its programs accredited; that's the Pennsylvania Power and Light Susquehanna Units 1 and 2.

I'd like to speak a few moments on recent utility experience in training. Back ten years ago, I was somewhat of a critic, I was heavily involved in training, I was a consultant, a licensing examiner for the Atomic Energy Commission and the Nuclear Regulatory Commission, examining reactor operators and senior reactor operators of utilities. I was a critic at that time of what I thought were weaknesses in training programs. There's no question that training of nuclear plant personnel is dramatically different today; it's better, and it's more comprehensive than it was at the time of the Three Mile Island accident. Training programs are rapidly becoming more systematic and performance-based, because the utilities are seeking to become accredited. They must be performance-based; they must be systematic. So there's a dramatic change taking place in the format of training programs. Improved training programs are being developed for personnel other than licensed reactor operators and SROs, and I mentioned the other eight positions and the possibility of others being added. I think the effect of having formal training programs for these other positions in the nuclear plant and the requirement that they have requalification training for those positions also, is going to have a dramatic effect in the next five to ten years on plant reliability in the United States. I think the key is competent maintenance personnel who are well trained in what they're doing. I think we'll cut down on the number of scrams, and the reliability will increase dramatically. More than having a dramatic effect in the licensed operator area, which we've been doing for years and doing a fairly reasonable job, the fact that we will now have formalized training and retraining of these other positions is going to have a dramatic long-term effect.

UTILITY EXPERIENCE

- The training of Nuclear plant personnel is dramatically different, better and more comprehensive than it was at the time of TMI-2 accident.

- Training programs are rapidly becoming more systematic and performance based.

- Improved training programs are being developed for personnel other than licensed ROs and SROs and STAs.

- Nuclear power plant simulators are recognized as one of the most effective training devices for operators. (also for maintenance).

- There are 44 of these multi-million dollar devices operating. An additional 24 are under construction or planned. (There were 10 simulators in operation in 1979).

- By the end of 1984, almost 1.6 million square feet of space (150,000 square meters) was dedicated to nuclear training (more than three times that in 1979).

- There were more than 2,100 full-time training personnel

  - an average of 24 instructors per plant

  - an average of 5 other training professionals per plant
    4 times as many as in 1979.

- In 1983, more than 4,500 people completed formal, initial training programs for 10 job categories

  - 43% increase over 1982.

- Today, virtually all plants have five or six shifts.

  - allows one shift to be retraining at all times
    (14-20% of time in retraining)

  - previously, four shifts was typical

- During 1983, more than 6,000 persons entered utility training programs for operator, technician, and maintenance positions.

- One utility having 3 nuclear sites indicated the following expansion of its nuclear training activities.

	1979	1985
•• Number of Nuclear Training Personnel	23	141
Training Budget	$1,055,000	$16,100,000

•• This utility has $37 million of capital investment in training facilities

•• In 1984 this utility had

- 637 people in operator training or retraining
- 893 people in craft training

• Congress, in the Nuclear Waste Policy Act, directed the NRC to promulgate regulations or other regulatory guidance for the training and qualification of nuclear power plant operators, supervisors, technicians and other appropriate personnel.

• In recognition of the INPO accreditation program, the Commission decided not to issue further regulations in the area of qualification and training.

• Instead it issued the Commission Policy Statement on Training and Qualification of Nuclear Power Plant Personnel in March 1985.

• The Policy Statement recognizes the industry effort through INPO and indicates that the Commission will not promulgate further regulations in this area for a period of at least two years.

• This cooperative approach by the NRC and the nuclear utility industry is generally viewed as an improvement in working relationships.

• However, there are some who doubt whether INPO will be successful, and whether INPO has the power to bring "weak sisters" into line in meeting the industry goal of seeking excellence.

• The onus is now upon the industry to demonstrate that this trust in their ability to improve the training of nuclear facility personnel is warranted.

Nuclear power plant simulators are now recognized as one of the most effective training devices for operators, and also for training instrumentation control technicians and maintenance personnel. Currently there are 44 of these multi-million dollar devices operating in the United States and an additional 24 of them are under construction or planned. There were ten simultors in operation in 1979, at the time of the Three Mile Island accident. By the end of 1984, there was almost 1.6 million square feet of space (about 150,000 sq. meters) dedicated to nuclear training, approximately three times as much as there was in 1979. At the end of 1984, there were more than 2,100 full-time training personnel working. That's an average of 24 training instructors per unit, and there's an average of five other training professionals at plants, people to help make slides, instructional material, people to help design curricula, and so forth. This number of training personnel is about four times as many as there were in 1979. In 1983, more than 4,500 people completed formal initial training programs for the ten job categories that I listed that INPO is accrediting. That's a 43% increase over the previous year of 1982. Today, virtually all plants have five or six shifts, where previously four shifts were typical. That allows one shift to be in retraining at all times. If you look at U.S. utilities today, you'll find that between 14 and 20% of operators' times are spent in retraining activities. During 1983, more than 6,000 persons entered utility training programs for operator, technician, and maintenance positions.

One utility, which has three separate nuclear sites, provided the following information. In 1979, they had a total of 23 nuclear training personnel. In 1985, they had 141. Their training budget in 1979 was just slightly over $1,000,000, and in 1985 it was slightly over $16,000,000. This same utility has a capital investment of $37,000,000 in training facilities, including simulators. Last year, the same utility had 637 people in operator training or retraining, and they had 893 craft positions in training. You can see that some of these utilities are running minor colleges or teaching institutions.

Congress recently passed the Nuclear Waste Policy Act. One of the things that they added in that act was a mandate to the Nuclear Regulatory

Commission to promulgate regulations or other regulatory guides, as they put it, for the training and qualification of nuclear power plant operators, supervisors, technicians, and other appropriate personnel. It was a case of Congress stepping in and telling the NRC, "You've got to improve training and do something about it." When Congress speaks, the NRC has to comply. This placed the Commission in quite a quandary because the INPO program was just getting started. The NRC started out to do just what Congress wanted them to do, which is to issue some regulations requiring some of these things that I've just talked about. The industry became concerned and upset and they organized themselves. I don't know if you've heard of NUMARC, that's the Nuclear Utility Management and Resources Committee; they organized themselves and spoke as one voice. They told the NRC commissioners that they were doing these things voluntarily through INPO. They were going much beyond the minimum requirements that Mr. Counsil was talking about on regulations. They were seeking to improve quality, seeking excellence, and they said that if the NRC then comes along and takes what they're doing voluntarily and makes this a requirement, the NRC will kill this effort they have through INPO. Well, it placed the Commission in quite a quandary, but I give them much credit. They decided that they would not issue regulations. They issued instead a Commission policy statement on training and qualification of nuclear power plant personnel. They issued it this past March. In that, they recognized the industry efforts through INPO, and they indicated that the Commission will not promulgate further regulations in the area of training and qualification for a period of at least two years. They're going to monitor it closely, and so forth, but said "We are not going to issue regulations." Congress appears to have accepted that. I have not heard any concern expressed. This, to me, is unique in U.S. nuclear regulatory history.

As I look at a number of other countries and their regulation, it is one of trust and working together to solve problems. That has not been our history, especially in the last ten years. Not because individuals did not want it, but because of our political and legal institutional problems, with many groups of Congress having oversight responsibility, some of which are pro-nuclear, some of which are anti-nuclear, some of which don't know much about nuclear. It has not been one of trust and mutual cooperation in solving problems. It's been just the opposite. It's been an adversarial relationship. However, there are encouraging signs that the current Commissioners are trying to work cooperatively with U.S. industry. There have been a couple of issues, this being the first one, which I think are significant. If it continues, it will also have a dramatic effect on the U.S. nuclear industry.

However, there's no question that there are a number of people, including at least one NRC commissioner, and most certainly members of the NRC staff, who are very doubtful that INPO will have a clout to pull this off and to be able to take the weak-sister utilities and bring them up to the level that is necessary. So there are people who are watching extremely closely. There are people who say that industry will never meet this commitment of having their programs ready for accreditation by the end of 1986. The onus is now upon the industry to demonstrate that it can do it. My own personal view is (I'm optimistic) that it's a worthwhile experiment. I'm highly encouraged by it, and I do think that, if it is successful, it's going to have a dramatic effect on the quality of personnel and, therefore, on reliability and safety of nuclear plants in this country. But time will tell. Does INPO have the clout? I believe they do. Will they bring that clout to bear? I'm encouraged. I think it's an exciting time at the moment to see the regulator and the regulated starting to work together in this country, as they did back in the earlier days under the Atomic Energy Commission. But only time will tell.

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2. Energy Reorganization Act of 1974.

3. Title 10, Chapter 1, Code of Federal Regulations - Energy
   Part 50, "Domestic Licensing of Production and Utilization Facilities"
   Part 55, "Operators'Licenses".

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VI U. S. EXPERIENCE IN KEY AREAS OF NUCLEAR SAFETY

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U. S. Nuclear Regulatory Commission

William J. Dircks
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U. S. Nuclear Regulatory Commission

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Basically, we view this discussion as an opportunity to describe a regulatory posture, or a regulatory environment, for the safe and economical use of nuclear power. The United States does not have a single nuclear or electrical power generation authority of any kind. The electrical power industry in the United States is very highly fragmented. The largest utility, perhaps, is the Tennessee Valley Authority, a federal government agency founded in the 1930s in order to bring electrical power to a region that was so undeveloped that it had virtually none, and it had no one to build it. That's how TVA was founded. It is the largest single utility that I know of. In other instances, you have utilities that serve only one community or one city. An example that comes to mind is the capital of the state of California, not far from here. Sacramento, California, is served by the Sacramento Municipal Utility District. It is a public corporation, you might say, a public body. They own one nuclear reactor, the Rancho Seco reactor, and I'd say that at least 50% of its power output is sold to other utilities, because it generates more than they need. So, in the United States, we are dealing with an industry which ranges from very large to very small, and many of the utilities are so small that they could not realistically buy a large power plant. As a result, there are difficulties in the U.S., because the owner and operator are not always one person, or even one very strong technical and financial group. However, in our system, we continue to believe that we should operate in this way. It means that the owners of the power plants have the primary responsibility for public health and safety, for protection of the environment, for safeguarding materials, for compliance with financial laws and economical performance, because that's their business. They're in the business of

SLIDE 1

CONFERENCE ON
MANAGEMENT OF NATIONAL NUCLEAR POWER PROGRAM FOR
ASSURED SAFETY

AN INTERNATIONAL PILOT COURSE

WILLIAM J. DIRCKS
EXECUTIVE DIRECTOR FOR OPERATIONS
U.S. NUCLEAR REGULATORY COMMISSION

SUBJECT: CONCEPTS OF NUCLEAR SAFETY - SAFE OPERATION OF NUCLEAR POWER SYSTEMS AND U.S. EXPERIENCE IN KEY AREAS OF NUCLEAR SAFETY

OBJECTIVE: HELP INTERNATIONAL LEADERS DEVELOP STRATEGIES AND MAKE DECISIONS ABOUT PLANNING AND ORGANIZATION OF THEIR PRESENT OR FUTURE NATIONAL SAFETY PROGRAMS AS THEY RELATE TO NUCLEAR ENERGY.

STANFORD UNIVERSITY
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producing or buying electricity and delivering it to the customers who use the electricity. Their regulation in the United States is both local and national. In simple terms, in the United States, if you are making electricity to sell directly to customers, like factories or home-owners, the regulation of how much you can charge is done by local authorities, usually at the state government level. If you are making electricity for sale to others, especially in other states, there is federal regulation of that by the Federal Energy Regulatory Commission, which has the responsibility for economic regulation or price. Now, given these circumstances, we in the NRC see the owner of the plant as having an economic function in electrical power, and coming to the appropriate federal safety agency to say, "Under what conditions will you authorize me to do this, to build a nuclear power plant and generate electricity?" So the responsibility is the owner's and never leaves the owner. It's a very important point, because when some of the owners get into nuclear safety and see how complex it can be, they have a tendency, especially the weaker ones, to turn to us and say, "Tell me what I have to do." And if you aren't careful, the regulator becomes responsible. After TMI, it is very interesting, the General Public Utilities Corporation filed a lawsuit against the NRC for letting them do it the wrong way and get into that accident. So, as regulators in your own countries, think of that. That responsibility must be clearly with the owner and the operator.

The government, the federal regulator, should provide assurance that this is done safely. That's the role. It's like a policeman, to observe that the laws are followed. Now, the very first point is a very good one to emphasize here. We have just talked about risk and how we are so much better able today to say what is safe enough. Minimum regulation for safety. Is it appropriate to say "as safe as possible"? I showed you the slides. The last slide I showed, that the GESSAR II risk estimate is now well below the range of debate with the safety goal. The Advisory Committee on Reactor Safeguards has asked me, "Why did you stop then? Why didn't you keep going and why didn't you further reduce the core-melt frequency? Why did you not further reduce the population

RESPONSIBILITY FOR SAFE AND ECONOMICAL NUCLEAR POWER PLANTS

INDUSTRY

(NUCLEAR FACILITY OWNERS, DESIGNERS, CONSTRUCTORS, OPERATORS AND VENDORS RESPONSIBILITIES AND COMMITMENTS)

- OWNERS HAVE PRIMARY RESPONSIBILITY FOR
  - PUBLIC AND WORKER HEALTH AND SAFETY
  - ENVIRONMENTAL PROTECTION
  - SAFEGUARDING IN THE INTEREST OF NATIONAL SECURITY
  - COMPLIANCE WITH FINANCIAL LAWS
  - ECONOMICAL PERFORMANCE

SLIDE 3

REGULATION OF NUCLEAR POWER PLANTS

GOVERNMENT

(TO ASSURE INDUSTRY MEETS ITS RESPONSIBILITIES AND COMMITMENTS)

- MINIMUM REGULATION FOR SAFETY
- REQUIREMENTS (STABLE AND PREDICTABLE, RISK BASED RULES, STANDARDS AND GUIDANCE CONFIRMED BY EXPERIENCE AND RESEARCH)
- LICENSING (STANDARDIZATION, ONE STEP, PUBLIC PARTICIPATION, DETERMINATION OF ACCEPTABLE RISK)
- INSPECTION (RISK BASED, ASSURE CURRENT AND CONTINUED ACCEPTABLE RISK)
- ENFORCEMENT (CORRECTIVE AND DETERRENT)
- INFORMATIVE (INFORMATION AVAILABLE FOR PUBLIC SCRUTINY)

The other part of interest is Part 55 of 10 CFR, which is entitled "Operators' Licenses", and in its entirety it refers to the qualification and training of personnel. In addition to these two parts of the federal regulations, the Nuclear Regulatory Commission has issued a number of different documents which provide guidance. These might appear in the form of what we call NUREG documents. In the Nuclear Regulatory Commission, they're called NUREGs, regulatory guides, etc. It's interesting that, although the guidance that's provided does not have the force of a rule or regulation, there's a continuing battle going on between people like Mr. Counsil, who's here this morning, and the staff, because the staff many times wants to take that guidance and impose it upon utilities saying, "You must do this." The intent of the guidance is to say, "This is one example of how you might implement the regulations, and if you implement them in this manner, it will be acceptable to the NRC. You can implement other ways, but you have to demonstrate why it meets the regulations." But many times the staff tries to impose the guidance as a regulation.

You will see in the first slide that there are only two categories of personnel that are licensed by the U.S. Nuclear Regulatory Commission. The first of these is the reactor operator, whom we refer to as the RO, defined as any individual who manipulates a control of the facility. By control of the facility, we mean any apparatus or mechanism which directly affects the reactivity or power level of a reactor. Now, when I make quotes from the regulations, I am not making a direct quote, I'm paraphrasing it for simplicity and, hopefully, for the ease of your understanding. The second type of person that is actually licensed by the Nuclear Regulatory Commission is the senior operator, SO, but in the field we refer to it as Senior Reactor Operator, SRO, much more frequently than we refer to it as SO. That's any individual who directs the licensed activitites of licensed reactor operators. Somebody who supervises the reactor operator in carrying out his licensed responsibility must have a license, and we call it an SRO license.

We now look at 10CFR50 and I'm directing your attention to the training aspects, and only for power reactors. There's information in there also for non-power reactors and other facilities, but I'm focusing these requirements which are stable, because they are based on risk and a wealth of experience and research. Now this is not to say that all of our present requirements meet this standard. I have previously mentioned the problems of charcoal filters and containment leak testing. We have many things we're looking at right now, and they are things that we can adjust to be more like this.

In our licensing approach, I should remind you that our laws prevent us from demanding standardization. We can only make it very favorable to standardize. We want one-step licensing. It is our practice in the U.S. to have extensive public participation, and that includes public hearings and so forth. We have no intention of changing that, and I think by streamlining or simplifying our license-review process, we can make that not the barrier that it has been with the old process. Our inspection program has changed very significantly in the last 10 or 15 years. We now have inspectors who understand what risk is and how one measures risk. Their work priority is often set by very specific risk considerations, and we apply our inspection resources much more intelligently. Our enforcement of violations is done the same way. We judge what's serious that way. And all of this, in our U.S. style, is available for the public to look at any time. We have in the U.S. many laws that are costly in some respects -- for example, the Freedom of Information Act, which enables virtually any member of the public to get virtually any document they want. These laws can be difficult but they are also very clear because the logic and the decision are open, and they're clearly arrived at.

So, here we give a summary of our experience: you have to have a commitment to excellence. You have to be determined to do it right, and I would say to do it right the first time. Both of us -- both the government and the industry. The government should be very careful about changing its requirements because, if you don't do it right the first time, the industry or the agency in your country which is trying to build the power plant cannot do it right the first time, because they don't know what the requirements are. Changing requirements are very difficult to deal with. So get it right the first time with a solid

SUMMARY OF U.S. NUCLEAR POWER REACTOR EXPERIENCE

A SAFE AND ECONOMICAL NUCLEAR POWER PLANT IS THE RESULT OF A GOVERNMENT AND INDUSTRY COMMITMENT TO EXCELLENCE IN PLANT

- SITING
- DESIGN
- CONSTRUCTION
- OPERATION
- SURVEILLANCE AND TESTING
- MAINTENANCE
- MODIFICATION
- DECOMMISSIONING

basis of regulation, carefully and strictly observed, and then the power company or the power agency can understand what those regulations are. If it is committed to excellence, and it must be, then siting, design, construction, operation, everything from the beginning to the end can be done effectively, safely, and economically. In the U.S. you get a lot of bad publicity for the very expensive plants. People don't realize that the difference in cost in nuclear power plants is tremendous in the United States. For every bad one that gets a headline, there are several good ones that cost 1/4 or 1/5 the level. The Shoreham Nuclear Plant is quite famous for its litigation, and hearings, and opposition, and everything else, and I think the Shoreham is now at something like $5,000 per kilowatt. That's the cost per kilowatt installed. It's about $5,000. And at the same time, there are power plants being licensed in this country which cost $1,500: Braidwood, Catawba. Not one, many! And part of it is just opposition. Those plants, though they might have opposition, do not have so many years of government and local opposition that have held them up. But they also have very effective management and people taking care to build, to test, and to operate those plants to a standard of excellence.

With respect to siting, I'd just like to repeat, because so many times, with foreign audiences in particular, I have had the comment raised that the U.S. siting standards are too strict. We have now done a very large body of work on nuclear power plant siting, both by plant type and plant design. We have considered very carefully severe accident prevention and mitigation, and the source terms. I see no reason at all to even consider making American siting standards stricter. If we had an incentive to do so in this country, we could even consider a much more lenient set of siting standards, but I don't see the incentive to do so. If we had public utilities trying to get reactor sites very close to cities, we might be able to justify that. And if we ever reach that point, I think we could. But until that time, we have very many sites just like the ones we have now, there's just no incentive. So I think it is a good assumption on your part that the old U.S. siting standards are the new U.S. siting standards -- the same.

NUCLEAR POWER PLANT SITING

- o PLANT TYPE AND DESIGN
  - SEVERE ACCIDENT PREVENTION AND MITIGATION
  - SOURCE TERM
  - ACCEPTABLE RISK
- o LOCATION
  - GEOSCIENCES
  - METEOROLOGY
  - ENVIRONMENTAL POOLING
  - DEMOGRAPHY
  - NATURAL EVENTS
- o EMERGENCY PREPAREDNESS/RESPONSE
  - EVACUATION
  - RECOVERY

As for location, we do have good standards now. Our risk analysis in things like seismic risk, earthquake risk, is much more difficult than what we call internal events -- for example, the loss of cooling and piping, and so forth. The earthquake-induced risk, the external event risk, is much more difficult to do quantitatively. It not only has great uncertainties, but they are different uncertainties. What is the earthquake hazard? How frequently should we expect a major earthquake? And even where we have comparatively a lot of data (you know there's a big earthquake fault right outside the back door here), we still don't have what I would call a level of confidence in those seismic risk analyses that would justify those risk analyses being on the same page with internal-event risk analyses. They're different sorts of things. But nevertheless, I think we <u>do</u> have good standards; they may be a little bit conservative in some respects. We have imposed combined-load requirements, assume a loss-of-coolant accident <u>and</u> an earthquake, things like that, that I think are unduly pessimistic. We're considering changes in those, but basically I think what we're doing, in geosciences and the other natural sciences, is confirming that what we have are good standards or unnecessarily stringent standards. In my staff right now, I have a group of meteorologists, and if you want to see them get mad, I threaten to break their rice bowl, and I say, "Do you know what? We don't need <u>any</u> meteorology data from a site. We don't need a big, tall tower. All we need is a windsock, a little wind-direction indicator, a weathervane, so that if you ever have an accident, a person puts his finger out there and says, 'Yeah, the wind is blowing that way and the weather looks bad.'" That's the only thing you can base a decision on. He can give you the last seven years of data from a 100-meter tower and it's a waste. It's not needed. And so, if anything, our meteorology is far more than enough. Our environmental monitoring we have carried to an international extreme. We've put cooling towers on Lake Erie. Demography, of course, is a changing thing, and it's reasonably well understood. Our population standard, I say again, 500 people per square mile is a typical value that we use for siting, and that's not very heavy population by most countries in comparison.

In emergency preparedness, which we associate with siting, we have gone like the swing of a pendulum. Before TMI, we had emergency preparedness, but we really didn't do much with it. And after TMI, everyone got screaming and hollering about how we needed emergency preparedness, and we developed the 10-mile planing zone, which many in the U.S. think is a 10-mile evacuation zone. We have gone to the other extreme, and now we're trying to bring some reasonableness to it. There are many requirements we have: the emergency off-site facility for utilities. We require utilities to have an emergency engineering center near the site. The Duke Power Company used their headquarters in Charlotte, North Carolina, as that center for McGuire, two units, and Catawba, two units. Oconee, with three units, is about 120 miles from Charlotte, down an express highway. They said they wanted to use the same energency facility. And we said, "Nope. You've got to build one close to the site." And they took us to court, they filed suit, and they lost, and they now have to build one next to the site. You want a private opinion? It's a waste of money. It doesn't help public health and safety very much at all. So when we look at emergency preparedness, expecially if you keep the core-melt probability low and keep the consequences of core melt low, then the effectiveness of having all of these arrangements is very questionable. So I think you will see in this next year, two years, in the U.S., a very significant reconsideration of emergency preparedness. It's very political. It was the way the Shoreham plant was blocked for the last three years by the local county. The county would not cooperate with the emergency preparedness and therefore stopped the plant. So it's very political, very difficult in the U.S., but I think you will see changes in that.

And last, the design of a safe and economical nuclear plant has to consider that you're going to operate it. And I again go back to what I said before. The owner's investment is hostage to the public health and safety. He's better off if that power plant faithfully and reliably generates electricity so that it is economically effective, and that makes it safety-effective. Therefore I'm grateful to say that we have this experience. We're now about to license our 100th reactor. But we

DESIGN OF A SAFE AND ECONOMICAL NUCLEAR POWER PLANT

- o RELIABILITY
- o OPERABILITY
- o SURVEILLABILITY
- o MAINTAINABILITY

  - APPLY EXPERIENCE AND RESEARCH
  - SIMPLICITY
  - RISK-BASED SYSTEMS APPROACH (SYSTEMS INTERACTIONS)
  - INTEGRATION AND BALANCING OF COMPETING RISKS
  - DESIGN-IN PROTECTION (HUMAN ERROR, FIRE, NATURAL EVENTS, SABOTAGE, HARDWARE FAILURE)
  - STANDARDIZATION
  - FAIL SAFE PHILOSOPHY
  - AUTOMATION
  - MAN-MACHINE INTERACTIONS (HUMAN FACTORS)
  - EASE OF SURVEILLANCE (SELF TESTING)

have to apply our thoughts and our efforts to reliability of operation, making sure that the plant is operable and not extremely difficult for the operator, so that not only does the machinery work reliably, the operator can reliably deal with it when the machinery stops or breaks or fails. And that includes the two things I have over on the side here, that you must be able to do surveillance, or monitoring, of the power plant, and it also includes good maintenance. There's a great deal of effort in our arena focusing on system realiability, operator reliability, and surveillance and maintenance. That, in effect, is the principal reason we're changing our very organization, to better apply our attention to these issues of safe operation.

VII THE U. S. NUCLEAR REGULATORY COMMISSION -- FUNCTION AND PROCESS

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THE U.S. NUCLEAR REGULATORY COMMISSION-FUNCTION AND PROCESS

Victor Stello, Jr.
Deputy Executive Director
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I had an opportunity to get a little bit of the flavor of what's been conveyed to you from some of the previous speakers this morning, and the impression I've gotten is that you've basically been hearing what the industry thinks of the NRC. I'm not going to bore you with what the NRC thinks of industry. I'm going to talk about the problems. I think all of you must be aware where you have a relationship of a regulator and a regulated there's a certain degree of animosity at times, a certain natural relationship that is probably very healthy in some respects in terms of disagreement and agreement and debate and discussion. I think that's very, very important in regulation. Regulation is very judgmental. We have developed, and you're going to hear an awful lot from me, and later in the week a lot from several other people, of some new ways to understand safety in terms of new techniques. All of you, I am sure, are aware of them, at least superficially, maybe some of you in detail. Probabilistic risk assessments give us the first opportunity to deal with safety issues in a quantitative sense and get away from what's been referred to as deterministic or judgmental kind of regulation.

An hour and a half this morning isn't going to give me a great deal of time to get into anything in very much detail. So I'm going to be moving very, very fast and covering a lot of topics. What I want to talk about briefly is the NRC philosophy of regulations. I think it's shared not only by NRC but everybody in the reactor safety business. Our mission is a very simple mission. Our mission is to assure somehow that the nuclear plants are operated, constructed and operated, safely. We have emphasized in the last several years an aspect of cost-benefit analysis which brings in another element of reliability -- availability. In terms of the impact of regulation, I'll be talking a lot about that later.

Our goals in terms of regulation and the goals of those who are regulated are not any different in most respects. A plant that is a very reliable plant, a plant whose capacity factor is very high, is a plant that is going to be very safe. So, from the utility point of view, if they can make these plants work very well, they indeed are going to be safe. They will have addressed and resolved all of the significant safety issues. A plant that has continuing problems with steam generators, or piping systems, or cracks, obviously won't be very reliable. Equipment that is not maintained very well obviously won't be very reliable. Those kinds of problems, when they come up, are of concern to the regulator. So what we're really interested in, and the whole reason the NRC exists is to deal with the safety issues. We have some other responsibilities to deal with environmental issues but that is not something I intend to touch on today.

I'm going to start by describing what the NRC is, how it's organized, how the licensing process works. Then I'm going to talk about some of the major contemporary issues that we face that I think will be common to the kinds of questions and concerns that you all have. Finally, I want to get into what I characterize as the decision making process. How do you decide to do something as a regulator? What do you consider? What is important in an issue and how do you decide an issue needs to be resolved via a fix, or whether it's a fairly unimportant issue. I'm going to be giving you three examples in some detail of the decision making process.

The first thing that we have to start with is: what is the NRC? The easiest way to describe the NRC is to show you an organization chart. You can see the top of the organization chart always represents where the bosses are. We don't have one; we have five. They're called commissioners. It is an interesting challenge to work for five people who very often have quite different views on how to approach an issue and resolve an issue. Underneath the Commission, the principal officer to the commission is the Executive Director for Operations, my boss. I'm his deputy. Underneath him are the major program officers, and I've not shown you an awful lot of the satellite officers. One that's important to you, for example, our Office of International Programs, is not shown on here. From your point of view, that's an important office, but from the programmatic, day-to-day activity

TOPIC OUTLINE

I INTRODUCTION

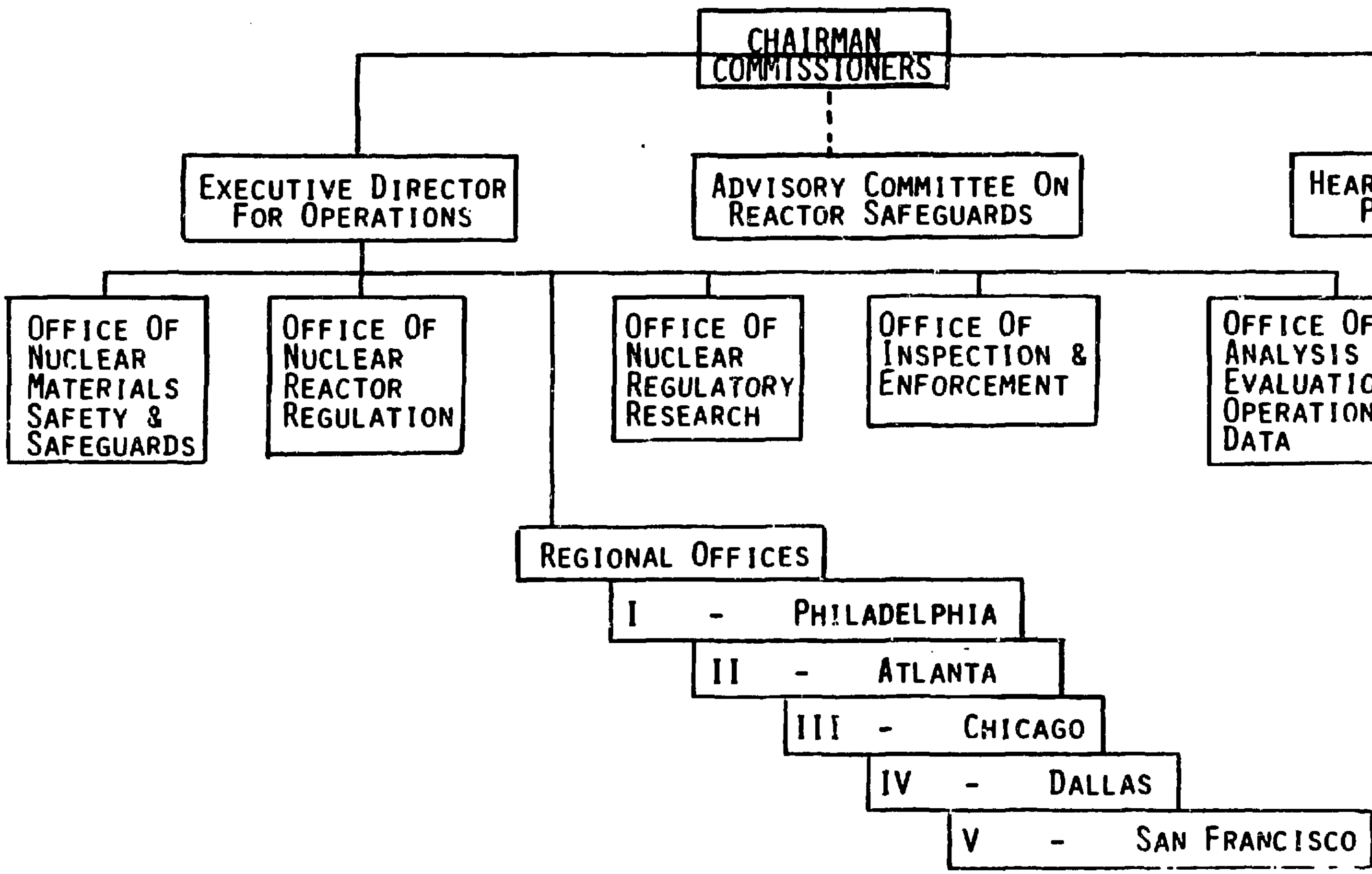
II U.S. NRC REGULATORY PHILOSOPHY
SPECIAL REGULATORY OBJECTIVES CONCERNING NUCLEAR POWER REACTORS
NRC ORGANIZATION
LICENSING PROCESS

III TWO MAJOR NRC EFFORTS TO MANAGE SAFETY IMPROVEMENTS (REQUIREMENTS)
COMMITTEE TO REVIEW GENERIC REQUIREMENTS (CRGR)
RULEMAKING AND GUIDANCE FOR BACKFITTING (GENERIC AND PLANT-SPECIFIC)
INDICATIONS OF SUCCESS

IV CONTEMPORARY REGULATORY ISSUES

V EXAMPLES ILLUSTRATING THE DECISION MAKING FUNCTION

ORGANIZATION OF
U.S. NUCLEAR REGULATORY COMMISSION



of the commission, they are not intimately involved in that activity. The principal offices in that regard are here: the Office of Nuclear Materials, Safety and Safeguards, the Office of Nuclear Reactor Regulation (NRR), the Office of Nuclear Regulatory Research, the Office of Inspection and Enforcement, the Office of Analysis and Evaluation of Operational Data (AEOD). I'll be talking a little bit about our hearing board panels. I'll mention the Advisory Committee on Reactor Safeguards. Then I'll be spending some time talking about our regional offices and their function.

Let's go to the slide and look at those offices. The Office of Nuclear Reactor Regulation; what's it for? Well, should an applicant want to build a nuclear power plant, someone within the agency has to decide if that power plant should be licensed. The function of NRR is to review that facility to determine whether it complies with <u>all</u> of our regulations, whether it has done so adequately and whether it is safe. Can we issue a license? So there's a licensing office. And that's its principal function, the licensing of nuclear power plants. There are a number of other functions, some of which are of interest to you. For example, licensing the operators who are at the controls of the power plant. How do you decide whether those individuals are or are not trained well enough to operate the plant? Decommissioning. At some point, these reactors will be decommissioned. It has been announced that three reactors in this country -- Humboldt Bay, Dresden, and Indian Point I will be decommissioned. After you've operated the plant and you wish to decommission it, someone has to decide if it has been decommissioned adequately. That's the same office, NRR.

<u>Research Office</u>. In the old AEC days Research did a lot of the very basic work evaluating the fundamental safety of plants. They did the early work on whether or not emergency core cooling systems functioned adequately. They did basic research on heat transfer in fuel rod bundles, finding out how the bundles behaved in the power plants. Fuel work. Materials work. They built and tested a number of very difficult transients in our LOFT facility and the Power Burst Facility, PBF. We actually took severely damaged fuel and measured the characteristics of fuel that was in a very advanced state of deterioration to get understanding of the safety

FUNCTIONS OF OFFICES

OFFICE OF NUCLEAR REACTOR REGULATION

- ISSUES AND AMENDS CONSTRUCTION PERMITS AND OPERATING LICENSES FOR POWER PLANTS AND FACILITIES OTHER THAN REPROCESSING PLANTS OR ENRICHMENT FACILITIES.

- DEVELOPS AND ADMINISTERS POLICY, REGULATIONS, AND PROCEDURES GOVERNING:

  - o LICENSING OF FACILITIES
  - o LICENSING OF OPERATORS
  - o STANDARDIZATION
  - o DECOMMISSIONING

FUNCTIONS OF OFFICES

OFFICE OF REGULATORY RESEARCH

- PLANS, RECOMMENDS AND EXECUTES RESEARCH IN
  - o ENGINEERING TECHNOLOGY
  - o RADWASTE MANAGEMENT, EARTH SCIENCE, HEALTH EFFECTS
  - o RISK ANALYSIS
  - o ACCIDENT EVALUATION - PREDICTING PLANT BEHAVIOR
  - o HUMAN FACTORS AND SAFEGUARDS
- DEVELOPS AND ISSUES REGULATORY GUIDANCE AND STANDARDS FOR REVIEW

of the plants. Research was the office that began, under WASH-1400, the very first of PRA work, Probabilistic Risk Assessment. Norm Rasmussen and Sol Levine began that work almost 12 years ago and developed this new way to gain an insight into reactor safety. Most of that work began in Research through our laboratories throughout the country and it is, in fact, continuing on there. They execute another function of the Commission. That's to prepare the rules and new standards, new Reg Guides, new detailed requirements to be imposed on the facilities.

The Office of Nuclear Material Safety and Safeguards. We have in excess of 10,000 material licensees in this country, who handle radioactive material of one type or another. You need to decide which of these licensees ought to be monitored or what safety requirements should be placed on them, depending on the kind of material they have and what they do. That is the particular function of our Office of Nuclear Material Safety and Safeguards. The safeguards end of it is an important subject because that deals with situations in which you have material which has the potential to be diverted and used for other than its intended purposes. The concern, obviously, is that such diverted highly enriched uranium or plutonium might be used to manufacture a weapon. This is a topic which is a source of considerable international tension and all of you are very familiar with it. This office is responsible for coordinating our programs overseas as well as setting forth the requirements for safeguarding material in this country.

Office of Inspection and Enforcement. Clearly, after you license a facility and they start to build it, you want to inspect it to make sure that it's built correctly. Hence, we have the office of inspection. What do they do? Well, a licensee says, "Here's how I'm going to build my facility," and we go out and inspect it to make sure it's built that way. After the facility is in operation you want to monitor it to make sure that it's operated safely. The inspectors do that. Most of this inspection activity is done through our regional offices, not through the office in Washington, D.C. Since TMI, we have emphasized another aspect of regulation which we had only done in a rather superficial way before TMI; that's in the area of emergency preparedness, both on-site and off-site. That is

<u>FUNCTIONS OF OFFICES</u>

<u>OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS</u>

- RESPONSIBLE FOR LICENSING AND REGULATION RELATED TO:

  o SAFEGUARDS ACTIVITIES

  o FACILITIES AND MATERIALS FOR TRANSPORT AND HANDLING OF NUCLEAR MATERIALS

  o DISPOSAL OF NUCLEAR WASTE

FUNCTIONS OF OFFICES

OFFICE OF INSPECTION AND ENFORCEMENT

- RESPONSIBLE FOR POLICIES AND PROGRAMS FOR INSPECTION AND ENFORCEMENT OF NRC REQUIREMENTS

  - o DEVELOP AND ADMINISTER INSPECTION PROGRAMS
  - o DEVELOP POLICY AND IMPLEMENT ENFORCEMENT ACTION PROGRAM
  - o DIRECT EMERGENCY PREPAREDNESS ACTIVITIES
  - o PROVIDE PROGRAM GUIDANCE TO REGIONAL OFFICES
  - o DEVELOP QUALITY ASSURANCE PROGRAM REQUIREMENTS
  - o PROVIDE TECHNICAL TRAINING FOR NRC INSPECTION PERSONNEL AND FOREIGN NATIONALS
  - o PROVIDE FACILITIES AND PROCEDURES FOR COORDINATED RESPONSE TO INCIDENTS

now a major activity in this country, the preparation for dealing with an emergency, both on-site and off-site. It means making sure that plans are developed to deal with emergencies, and that training necessary for the people that respond to emergencies is conducted throughout the year. As I understand, next week you'll be going to Chattanooga to our training facility. That facility is also under our Office of Inspection and Enforcement.

After TMI, there was a question of whether we were doing an adequate job of looking at the experience from our operating reactors, looking at where might we have a problem, why did we have that problem, how was that problem corrected, or was it corrected. We have a great deal of data being generated every day on things that don't go as well as they should in a facility, or a piece of equipment malfunctions, or an operator makes an error. How can you take that information, evaluate it, trend it, understand what it is that went wrong, and how it ought to be corrected. That basic function, was assigned to this office, which I refer to as AEOD. Their job is to look at the experiences that we have had, what we have learned from that experience, and what we ought to do about it.

Now our regional offices - There are two particular programs in the regional offices that I want to spend at least a moment on. I have already told you that's where the principal function of inspection is conducted. One of those programs is to develop a system of what we call our resident inspectors. Each operating plant has two resident inspectors who live and work at that particular plant. That's where they spend their normal work days and they come in, typically, during 8-5 working hours. But they also go during weekends and they spot check on the back shift to observe how the plant is operated around the clock. So our resident program is now a very large one in the agency. They also have inspectors who are trained in metallurgy, NDE work (nondestructive examination work) who are experts in health physics, who are experts in reactor physics, who are experts in mechanical equipment, and these experts also do inspections out of each of our five regional offices. The closest one to us here today is at Walnut Creek, near San Francisco. That's where these regional inspectors normally stay and then they go and visit facilities periodically to do these inspections.

FUNCTIONS OF OFFICES

OFFICE OF ANALYSIS AND EVALUATION OF OPERATIONAL DATA

- RESPONSIBLE FOR ANALYSIS AND EVALUATION OF OPERATIONAL SAFETY DATA FROM ALL NRC-LICENSED ACTIVITIES
- COORDINATE DATA COLLECTION, STORAGE, AND RETRIEVAL
- MAKE EVALUATIONS AVAILABLE TO OTHER NRC OFFICES AND TO REGULATED INDUSTRY
- VERIFY THAT ACTION HAS BEEN TAKEN
- ASSESS EFFECTIVENESS OF PROGRAM
- INTERACT WITH OUTSIDE ORGANIZATIONS HAVING SIMILAR OBJECTIVES
- REPORT PERIODICALLY TO U.S. CONGRESS

FUNCTIONS OF OFFICES

REGIONAL OFFICES

1. RESPONSIBLE TO CONDUCT THE INSPECTION AND ENFORCEMENT PROGRAMS OF THE AGENCY

2. MAINTAIN A CONTINUING EFFECTIVE INSPECTION PRESENCE BY THE ASSIGNMENT OF RESIDENT INSPECTORS ON-SITE AT EACH REACTOR PLANT IN THE COUNTRY, INCLUDING BOTH OPERATING PLANTS AND PLANTS UNDER CONSTRUCTION

3. PERIODICALLY PREPARE WRITTEN INSPECTION REPORTS FOR EACH PLANT THAT, OVER APPROXIMATELY 1 YEARS TIME, CAN BE USED TO ASSESS THE UTILITY OWNER'S OVERALL PERFORMANCE IN MAINTAINING SAFE OPERATIONAL CONDITIONS AND PRACTICES.

4. CONDUCT MANY SPECIAL AND ROUTINE INSPECTIONS TO ASSESS PLANT OWNER'S PERFORMANCE WITH RESPECT TO NRC REQUIREMENTS IN CONSTRUCTION OF NEW PLANT, MODIFYING EXISTING PLANT, AND ESTABLISHING ORGANIZATION AND PROCEDURES AND PERSONNEL STAFFING AND TRAINING.

REGIONAL OFFICES (CONTINUED)

5. PREPARE AND ADMINISTER EXAMINATIONS TO LICENSED OPERATING PERSONNEL, AND ISSUE LICENSES

6. PROVIDE THE FIRST POINT OF CONTACT FOR THE NRC IN THE EVENT OF A PLANT ABNORMAL INCIDENT OR EMERGENCY

7. IN THE EVENT OF IDENTIFIED VIOLATIONS OF NRC REGULATIONS, PROVIDE THE FIRST LEVEL INVESTIGATION AND EVALUATION OF CIRCUMSTANCES THAT MAY LEAD TO THE ASSESSMENT OF A CIVIL PENALTY ON THE PLANT OWNER.

Another program that we have developed over the last ten years is what we call our SALT program -- Systematic Assessment of Licensee Performance. What does that mean? In simple terms, when you have nearly a hundred plants in operation, as we have today, and almost 30 under construction, you're interested to know which particular licensees, or which plants, are not performing as well as they ought to, and why? You can check off the kinds of things you are interested in. How well do they do in maintenance, for example? Or quality assurance? Or training? Or health physics? And you send in inspectors throughout the year. Then, at the end of the year, you get all of these people who have been out there inspecting these plants, and you bring them all together and you say, "Well, now, let's sit down and talk about how well did this particular licensee do during this past year? What has he done well, and what has he done poorly? When he's done something poorly, let's recognize that and develop an inspection program to try to get that problem corrected. On the other hand, where he has done well, then there's no point in us going out and doing a lot more inspection in that particular area, because he's doing it well. Let's take our resources in that area and find a facility where it's not going well and use them there." In fact, I was surprised to learn just last week that a particular utility, one of our best performing utilities in this country incidentally, takes the SALT reports that we write and uses them to decide who gets bonuses in their management system. Now I'm not sure we ever envisioned our reports to be used for that purpose, but it is interesting. Here's a plant doing very, very well, and they're serious. They look at how the NRC views what they've done and if they get a poor grade in some area, that particular manager doesn't get the kind of a bonus he might have liked.

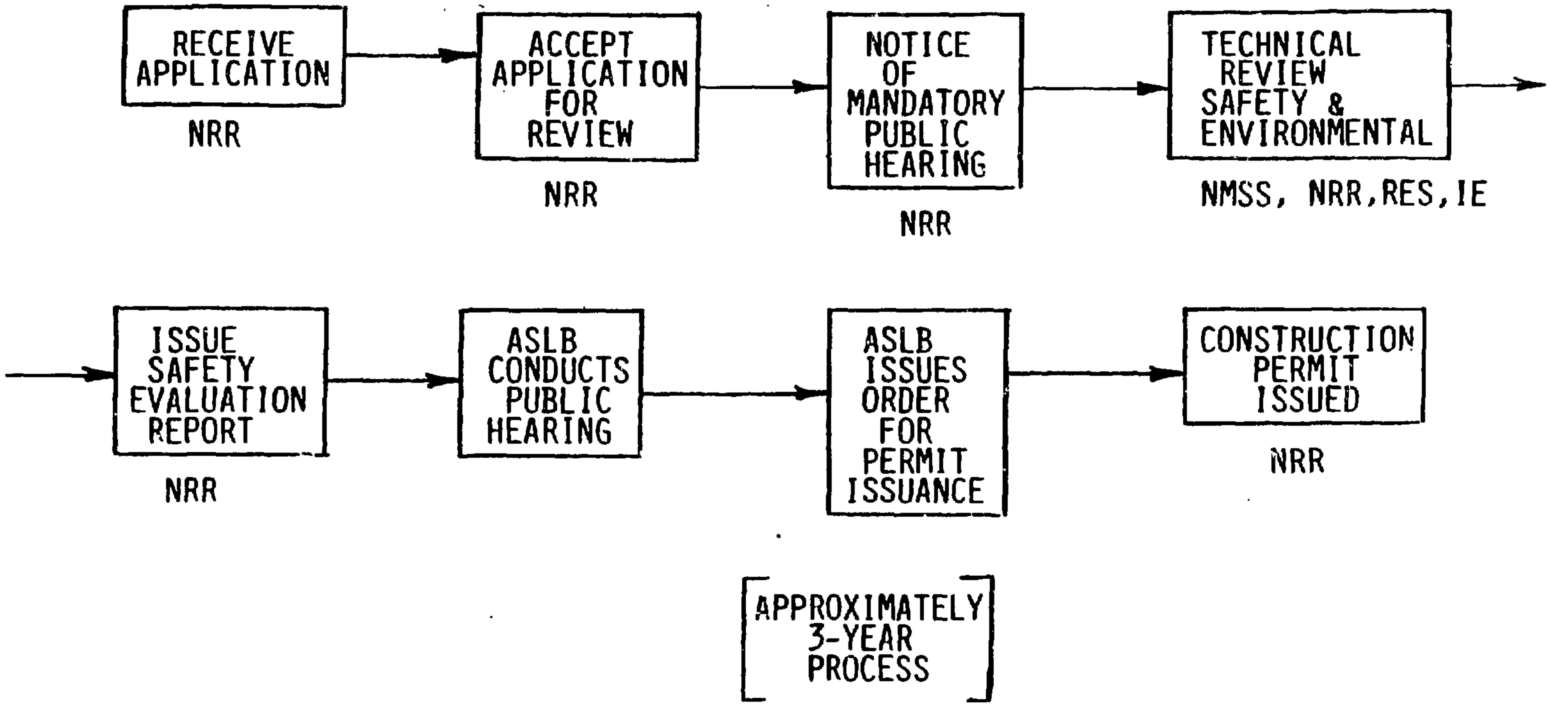
The last thing I want to touch on regarding the regional offices: If ever you have an incident in a plant, you're going to need to respond and respond quickly. We have response teams that have been set up, and arrangements made, so that we can take from each of our regional offices our experts and get them out to any particular site in the event of an incident. We have a whole system for communicating that through the regional offices with a large operations center in Washington, and if any of you happen to be in Washington, maybe you can come by and see it. It is there we collect the senior managers, including the Chairman of the Commission, in a room to manage the NRC's responsibilities during an incident.

There's two more charts that I want to go over. I want to describe for you the licensing process. No one from industry will believe that it is this simple. But look, all you have to do is give us an application and everything just clicks right through and it all works very smoothly. If the truth be known, I guess some of the plants in this country have taken 14-15 years from the beginning of that process until the end of the next slide when they are allowed to operate. Some of them have done much better. One has done it in 60 months.

What is the licensing process? - First, you have an application that's reviewed by NRR. That's what I was talking about, the technical review to decide everything is okay. They finally decide everything they need is in the application and accepted. Then we have to give notice of the next step. We will require (it's not optional) a hearing at some point in this process. The process of technical review sometimes can last several years. All of the details in the application are reviewed to determine if they comply with all of our regulations. This stage is concluded with the issuance of the Safety Evaluation Report (SER). If we disagree with the licensee on some point, our SER will say the licensee proposed to do x, and we don't agree that x is okay; we want y. At that point the hearing would start, and we'd go into the hearing process and tell the hearing board that we don't agree with the licensee in certain areas but in the other areas we do agree with him and we think it's okay to issue a construction permit. After that hearing process is complete, the board allows the issuance of the construction permit and then begins that inspection process. After the licensee gets a construction permit and builds the plant he wants to operate the plant. So what he has to do is do it all over again. He submits a new application and we go through the whole thing. The only difference being, essentially, that now there's issued what is called a "Notice of Opportunity for Hearing", which means the hearing isn't mandatory; it is just an opportunity. Based on

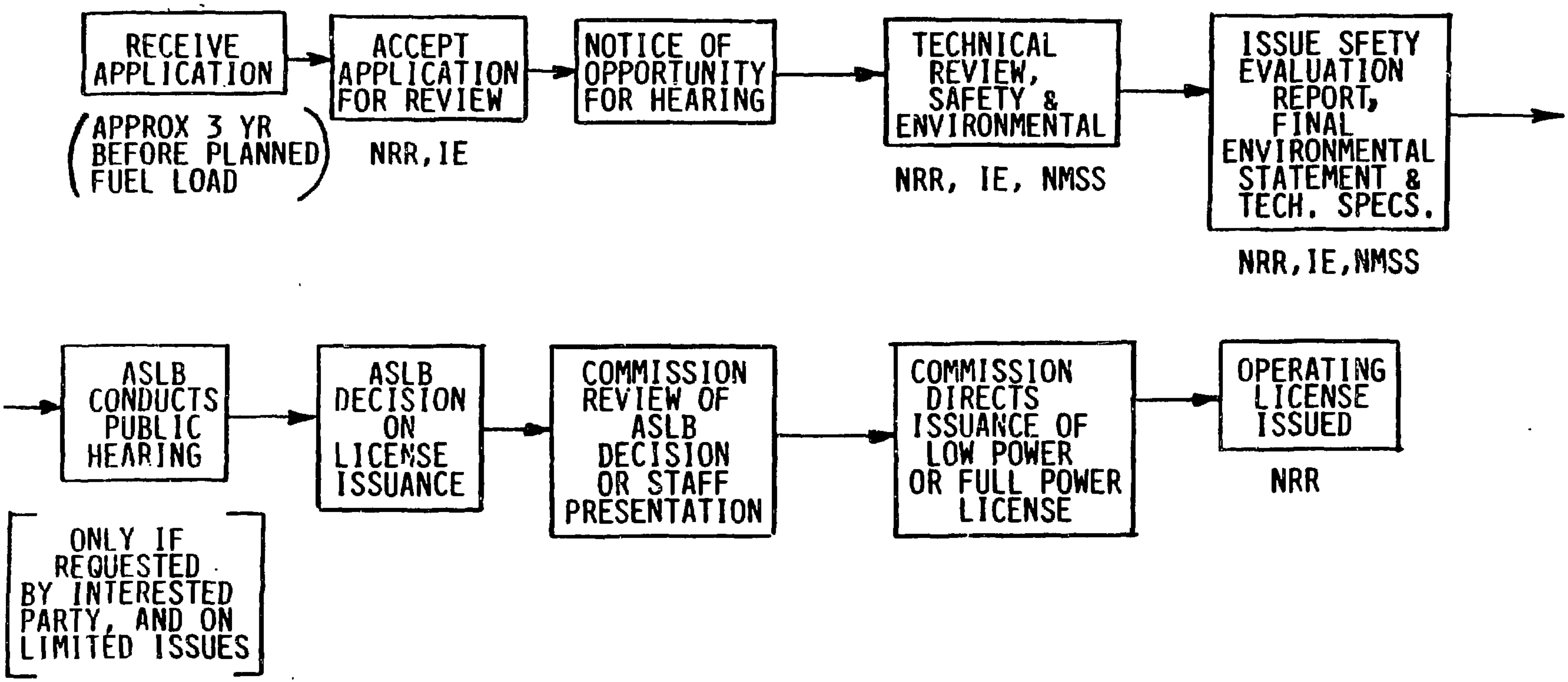
TYPICAL LICENSING PROCESS

CONSTRUCTION PERMIT



TYPICAL LICENSING PROCESS

OPERATING LICENSE



experience, however, it's the same as mandatory because just about every time we issue a notice, we have a hearing. So essentially the process is repeated before the licensee is allowed to operate the plant.

I have been asked my view of this process, whether I would like it to continue or would prefer to see it changed. It ought to be changed. There is a proposal to change it. It's part of our licensing reform package. What's the logical answer for avoiding this kind of a problem? It's called "standardization" and it reduces licensing to one step. When any particular applicant wants to build a plant, he comes to the NRC and says, "Look, I have this power plant design. I want you to give me a certificate that says I met all of your requirements. I'll give you all of my detailed design work and I want you to tell me that, if this plant is in fact constructed as I say, it can be operated. Called one-step licensing. When the NRC is finished, they will take a standard plant through that long hearing process and then incorporate it into the rules so that any applicant who wants to build this plant in the future need only say he's going to build that plant. There would no longer be any issue in terms of the adequacy of those details of design. Called one-step licensing. To take that one step further, you can also have applicants who want to build plants come to the NRC and say, "Look, we have this wonderful site that we want to build a plant on. Would you look at this site and approve this site as an acceptable site for building a plant?" Then, when the applicant wants to build it, all he has to do is say, "I'm going to use one of those sites which are now pre-approved by the NRC. I'm going to use one of these plants which the NRC has incorporated in its regulations saying, 'It's okay.'" And then he goes to a hearing and what are the only issues that are left for the hearing? There'll be detailed emergency planning for the state and locale where he is, and then the competency of that particular utility, the strength of that utility to go ahead with this operation. Those will be the only issues. There'll be no more details of design and all of that. Licensing reforms, new legislation that'll make that happen, is now being considered by our Congress. In airplanes there is an analogy - an airworthiness certificate. A manufacturer gets an airplane, and he flies it and shows it's okay, and the FAA issues an airworthiness

STANDARDIZATION

1. A POLICY STATEMENT WAS ISSUED IN 1978 AND STANDARDIZED DESIGN APPROVALS WERE ESTABLISHED IN OUR REGULATIONS.

2. THE COMMISSION'S SEVERE ACCIDENT POLICY TO BE ISSUED SOON, ADDRESSES STANDARDIZED DESIGNS.

3. AS ONE OUTCOME OF NRC LICENSING REFORM EFFORTS, THE PROPOSED NUCLEAR POWERPLANT LICENSING AND STANDDARDIZATION ACT WAS SUBMITTED TO CONGRESS IN FEBRUARY 1985. CONGRESS HAS NOT ACTED ON THAT LEGISLATION.

4. NRC IS DEVELOPING A REVISED POLICY ON STANDARDIZATION. INDUSTRY VIEWS (AIF) ARE BEING SOUGHT ON THE DIRECTION THAT STANDARDIZATION POLICY SHOULD TAKE.

LICENSING REFORM

1. THE COMMISSION WANTS TO DEVELOP ANDIMPLEMENT CHANGES IN THE LICENSING REGULATIONS TO (1) BENEFIT HEALTH AND SAFETY AND (2) PERMIT A MORE EFFECTIVE AND EFFICIENT LICENSING AND INSPECTION PROCESS.

2. BACKFIT MANAGEMENT, CHANGES IN FORMAL PUBLIC HEARING PROCESS, AND STANDARDIZATION SHOULD PROVIDE GREATER STABILITY IN LICENSING STANDARDS AND CRITERIA FOR APPROVED DESIGNS, AS WELL AS MORE EFFECTIVE AND EFFICIENT UTILITZATION OF NRC AND PUBLIC RESOURCES.

LICENSING REFORM (CONTINUED)

3. IN FEBRUARY 1985, THE NUCLEAR POWERPLANT STANDARDIZATION ACT OF 1985 WAS INTRODUCED IN THE U.S. HOUSE OF REPRESENTATIVES (H.R. 1029)

4. PURPOSES OF THE PROPOSED LAW ARE:

   A. TO FACILITATE USE OF PREAPPROVED SITES AND STANDARDIZED DESIGNS.

   B. TO PROVIDE FOR ISSUANCE OF ONE LICENSE TO BOTH CONSTRUCT AND OPERATE A PLANT.

   C. TO IMPROVE THE STABILITY OF LICENSING STANDARDS, CRITERIA FOR NUCLEAR POWERPLANTS, AND PRIOR NRC LICENSING APPROVALS.

<u>LICENSING REFORM</u> (CONTINUED)

5. IMPORTANT FEATURES OF HR 1029:

   A. EARLY SITE APPROVAL - APPROVED FOR 10 YEARS, MAY BE RENEWED FOR ADDITIONAL 10-YEAR PERIOD.

   B. CONSTRUCTION AND OPERATING LICENSES - PROVIDES FOR SINGLE STEP LICENSING, WITH OPPORTUNITY FOR PUBLIC HEARING 9 MONTHS OR MORE BEFORE OPERATION.

   C. CHANGES TO ATOMIC ENERGY ACT OF 1954 TO SET CRITERIA GOVERNING ALL ADDITIONS DELETIONS AND MODIFICATIONS TO COMMISSION REGULATORY REQUIREMENTS - WOULD REQUIRE CENTRALIZED, SYSTEMATIC, DOCUMENTED REVIEW OF THE PROPOSED IMPROVEMENTS IN SAFETY, AS WELL AS ESTIMATES OF COST OR OTHER IMPACTS.

certificate and says, "Go ahead and build them." That's the kind of process we're looking for.

Let me move on now to another important subject, the management of accidents beyond the design basis. Think a little bit about Three Mile Island and that accident. What was the Three Mile Island accident? If you look at our regulations, there are accidents that are called "Classes 1 through 8", and then you have what we call a "Class 9 accident". What does that mean? It means that something other than Class 1 through 8. What does it typically mean? It means that you haven't melted down the core. The core is intact; it's not significantly damaged. What happens when you go beyond that? You're what we call "beyond design basis." Typically, you are going to approach core-meltdown accident. There's a lot of things about TMI. TMI did not melt down, but clearly TMI was beyond the design basis. If you look at some of the TV cameras they put inside of the core, you'll see that there's half of it that ain't there. So half of the core was damaged so badly it's not intact. So it was clearly a significant and substantial accident. There's a lot of things we learn as we look at how they responded to that accident. I think they are very important. Let me touch on at least a couple of these. You should think about these a lot. Before the accident, all the procedures in the plant were what we call "event-oriented". That means you had to know what happened in the plant. Did I have a steam-generator tube rupture? Did I have a leak in a pipe, leading to a loss-of-coolant accident? Did I have a break in the steam line someplace, in the PWR? Did I have a stuck-open valve? What was going on? Then you had the procedures to respond to the event. At TMI, that got them into some difficulty; the thinking that you can decide beforehand what event is actually going on. So what you want to do is get away from that. Go to what we call "symptom-based procedures". That is, if the water level is going down in the reactor, and I know that's going on, I don't care why it's going down. What can I do to stop it from going down and make it go up? Respond to those symptoms, rather than events. It's called system-oriented procedures. All of the reactors in the United States are in the process of converting over from event-oriented procedures to system-oriented procedures. Those who have not put reactors in operation,

MANAGEMENT OF ACCIDENTS BEYOND DESIGN BASIS

1. DEVELOPED SYMPTOM-BASED EMERGENCY OPERATING PROCEDURES FOR EACH PLANT, TO COPE WITH OR AVOID DEGRADED CORE ACCIDENTS.

2. IMPROVED INSTRUMENTATION AND DISPLAYS TO PROVIDE OPERATORS WITH IMPORTANT INFORMATION.

3. IMPROVED HYDROGEN CONTROL FOR PRESSURE SUPPRESSION CONTAINMENT DESIGNS.

4. HUMAN FACTORS IMPROVEMENTS IN CONTROL ROOMS.

5. IMPROVED TRAINING AND TESTING OF OPERATORS, ESPECIALLY IN USE OF EMERGENCY PROCEDURES.

6. PERIODIC EMERGENCY RESPONSE EXERCISES, ON SITE AND OFF SITE.

7. IMPROVED COMMUNICATIONS AND INFORMATION SUPPORT TO ACCIDENT MANAGERS - EOF, TSC.

or countries that are planning them, this is the way that I think you want to go.

The instrumentation - Wouldn't it have been wonderful if at TMI they just had some simple things in the control room, like maybe a clock in front of them to show how the pressure and temperature responded as a function of time? Was there adequate instrumentation in the control room, or wasn't there? I think there's a consensus that, based on what we found from TMI, there were a number of places that instrumentation needed to be improved. The kinds of instrumentation that needed to be improved were identified and documented and issued as a Reg Guide of the NRC. The number of that Reg Guide is Reg Guide No. 1.97, and it identifies the kind of instrumentation that you want to pay attention to.

Another result of TMI is: No more discussion about whether you can get enough hydrogen to create a problem. No more. Hydrogen in a reactor is an important consideration so the need for hydrogen control for both PWRs and pressure suppression BWRs is evident. There's no question that you can get it.

Human factors engineering in the control room - Handles that were in the wrong places, buttons that were wrong, gauges that that were too far away to be read. There were a lot of fairly obvious things wrong in the control room. We have some techniques now to ask the question, "How do you design a control room, so that you've done a good job, recognizing what the operator has to work from that control room?" Do it up front. Where most of our plants are already built, that's difficult to do, but there's still a great deal that you can do to identify where the human error deficiencies in the design are, and there's a great deal that you can do to fix it. Just recognizing it sometimes is a fix, knowing that it's a problem.

What you've heard most talked about TMI is probably summed up in terms of inadequate training. I coined a phrase to describe what I considered to be the principal phenomenon at TMI, the mindset of the operators. They came out of a training program, and they were trained, by God, to look at that pressurizer water level, and that's it. And if that pressurizer water level is okay, you're okay. So they had a mindset in terms of how to respond. And I remember one point in the incident telling them, "Look, if you get superheat in your primary coolant pipes, your core's uncovered. It doesn't matter what your pressurizer is doing. You can't get superheat unless you uncover the core. The thermodynamics are such that you just can't do that." But I couldn't make them understand it. They had a mind set that, if the pressurizer level was up, the core was covered with water and that's okay.

Training. Make sure that the operators have training in some of the fundamentals of heat transfer, some of the fundamentals necessary to cope with these kinds of emergencies. In fact, TMI raised a controversy over whether we ought to have college-graduated engineers as Senior Operators. The controversy has, in fact, never stopped. Immediately after the accident we required them to have an engineer in the control room and we called them shift technical advisors. The controversy still goes on, however.

Source term. Everyone is aware that there has been a great deal of effort. It's probably the largest program that we have had on an international scale in terms of developing the kind of research and information and knowledge necessary to reassess the source term. That process is ongoing now. There has been a new suite of codes developed and they've been published. There's a report, BMI NUREG 0956. It was published I think just a week or two ago, which describes the summation of all this. It has had intense peer review. The American Physical Society put together a committee to review it, and their report has been issued. And things are encouraging. It's probably what you would expect, that the source terms that have been used in reactor safety analysis are too conservative. That's what everybody believed and the results of the research work are confirming them. What I've done in the chart is tabulate for you the old source terms which show the number of early fatalities per reactor year Remember how this is done -- how many fatalities there are times the probability. You see on the slide the numbers based on WASH-1400 source terms and then those based on the source terms that we have developed as a result of that research work.

NRC SOURCE TERM WORK

1. IN 1983 STARTED TO REASSESS TECHNICAL BASES FOR ESTIMATING SOURCE TERMS FOR POSTULATED SEVERE ACCIDENTS.

2. STUDIED QUALITY, TIMING, AND CHARACTERISTICS OF RADIONUCLIDE RELEASES TO ENVIRONMENT.

3. HEAVILY INVOLVED NATIONAL LABORATORIES, UNIVERSITIES, AND INDUSTRY IN A MULTI-NATIONAL DEVELOPMENT EFFORT. NOW HAVE A SUITE OF CODES FOR SOURCE TERM PREDICTIONS - A MAJOR ADVANCEMENT. CODES ARE COMPLEX AND REQUIRED SKILLED USERS.

4. NRC NOW STUDYING PUBLIC RISK IMPLICATIONS OF DEVELOPMENT WORK RESULTS.

5. LARGEST SINGLE FACTOR IS CONTAINMENT PERFORMANCE GIVEN CORE DAMAGE/MELT.

6. CANNOT GENERALIZE OR MAKE GENERIC CONCLUSIONS AT THIS TIME.

7. PREVIOUS SAFETY REQUIREMENTS MAY NEED REEXAMINATION.

Updated Risk Estimates for the Surry Plant Using WASH-1400 Accident Frequencies

Analytical Method	Early Fatalities (per reactor year)	Latent Fatalities (per reactor year)
WASH-1400 Source Terms WASH-1400 Containment Evaluation	4.0×10^{-5}	1.6×10^{-2}
BMI-2104 Source Terms WASH-1400 Containment Evaluation	1.1×10^{-5}	6.7×10^{-3}
BMI-2104 Source Terms Containment Reevaluation	3.1×10^{-6}	3.4×10^{-3}

- **Sources of Uncertainty**
  - Event Frequencies
  - Source Term Analytical Procedures
  - Containment Behavior
  - Consequence Calculations
- **Uncertainties Will Be Taken Into Account in NUREG-1150**

On the first line we have the WASH-1400 source terms with the WASH-1400 containment. The next line is just new source term work without dealing with any changes to the containment. And the third line is changes to the source term plus the containment. What we found in doing a lot of the research work was that not only are the fission products that come out significantly different than we thought but the performance of the containment is much, much better than we thought it was. In fact, the American Physical Society review says that, in a PWR, they consider containment failure to be unlikely. That is, even if you have a meltdown in a reactor, in a PWR, it's unlikely that you're going to get a failure of the containment due to things like steam explosions. Early containment failure. You can get a containment failure later by overpressurizing it, simply because you don't have heat-removal capability but it takes a long, long time and hence there are a lot of natural mechanisms to get rid of the fission products while all that's going on. This will continue to be a subject for considerable discussion for quite some time.

Now, I want to touch on another issue that I think can be summarized very simply -- safety goals. What is a safety goal? I view a safety goal in simple terms as the Commission's statement of how to use probabilistic risk assessment in regulation. That's where the safety goal really is. It gives you goals in terms of how they expect reactors to be operated. It goes further to give them in quantitative terms: there shall be a core-melt frequency as a goal of 10^{-4}. Then they go on to say that the risk should not be more than one-tenth of a percent of the risk from all accidents that people are subjected to. Car accidents, falling off ladders, lightning. Power plants should not add more than a tenth of a percent to the risk from all of those others of you having a fatal accident. In addition, it adds, one more goal, that operation of a reactor should not cause more than a tenth of a percent increase in cancer deaths. Those are long-term, over 30 years. The only way to get numbers that you can measure against those goals is by going to PRA. That's why I say I consider the promulgation of a safety goal as a policy of the Commission to be a simple statement of how it is that the Commission wants PRA used. A lot of controversy exists over safety goals. Is 10^{-4} low enough? Should it be

SAFETY GOALS

1. A HIGH LEVEL NRC TASK GROUP HAS MADE ITS REPORT TO THE COMMISSION ENDORSING THE ADOPTION AND ISSUANCE OF FORMAL SAFETY GOALS.

2. THE GOALS PROVIDE TARGET LIMITS ON OFF SITE PUBLIC HEALTH EFFECTS AND ON EXPECTED VALUE OF CORE MELT FREQUENCY.

3. THERE IS CONTROVERSY OVER THE CORE MELT FREQUENCY VALUE, AND OVER THE DESIGN AND USE OF COST/BENEFIT ANALYSIS TO JUSTIFY THE IMPOSITION OF NEW REQUIREMENTS.

4. THE GOALS, WHEN ESTABLISHED, WILL AID BUT NOT REPLACE THE METHODS USED NOW TO ESTABLISH NEW REQUIREMENTS.

5. IT IS EXPECTED THAT THE COMMISSION WILL DECIDE ON THIS TOPIC IN THE NEXT FEW MONTHS.

10^{-5}? Should the 10^{-4} mean a full-scale core meltdown or should it just mean severe core damage? One of the shortcomings of PRA is that you can't tell the difference between a full-scale core meltdown and one that's going to just give you core damage. There were a lot of ways to stop TMI's accident. If you had calculated what was going on, you would have said, "TMI is going to melt down. It's going to be a full-scale core meltdown." But it wasn't. There are ways to stop a very severely damaged core, very severely damaged, and prevent it from melting down. But the PRA techniques just aren't mature enough to be able to make those kinds of distinctions. Nevertheless, that's what a safety goal is. A safety goal is a way in which to describe how the Commission wants the industry to use this new-found technology called PRA.

I would now like to turn to the subject of decision making in regulatory matters. All of you are going to be faced with making decisions of one kind or another. How ought you to go about making these decisions? Rather than generalize, I decided the easier way to do that is to take a couple of examples and think through them together. A couple of issues that have come up. The first one that I want to talk about is station blackout. What is a station blackout? Even here in the United States, with the interconnections we have, we have data which show clearly that, from time to time, nuclear power plants lose off-site power. The ability to use the off-site network to drive the pumps and equipment in the plant is lost. The natural question that gets raised is: "Is the on-site power system, the normal AC power system, reliable enough in view of the frequency of off-site power loss, to ensure that the public risk is low enough?" A related consideration is that, even if you lose all off-site power and on-site power, we know that nuclear plants with just DC power, battery power, can operate for a significant length of time. How do you put all that together and try to answer some questions. Do we need to change anything with facilities? Well, you're going to get a number of questions coming to mind pretty quickly. How are the diesel generators arranged in the plant? Is it one out of two, two out of three that we can expect to operation if called on? What's their reliability? Has the maintenance of the diesel generators been poor so their reliability is low?

ITEM: A-44 STATION BLACKOUT

ISSUE

- ARE THE NORMAL AND EMERGENCY AC POWER SOURCES RELIABLE ENOUGH TO ENSURE LOW PUBLIC RISK FROM A SEVERE CORE DAMAGE ACCIDENT?

- SHOULD ALL PLANTS BE REQUIRED TO TOLERATE A TOTAL LOSS OF AC POWER, AND IF SO, FOR HOW LONG?

CONSIDERATION IN DECISIONMAKING

- o IMPACT OF DIESEL GENERATOR CONFIGURATION
- o DIESEL GENERATOR RELIABILITY
- o LOSS OF OFFSITE POWER SUSCEPTIBILITY
  - FREQUENCY AND DURATION OF PLANT CAUSED LOSS
  - FREQUENCY AND DURATION OF GRID CAUSED LOSS
  - FREQUENCY AND DURATION OF WEATHER CAUSED LOSS
- o CAPABILITY OF PLANT TO COPE WITH LOSS OF AC POWER
  - DECAY HEAT REMOVAL SYSTEM DEPENDENCY ON AC
  - DC POWER FEATURES
  - COOLING OF PUMP SEALS
  - CONTAINMENT HEAT REMOVAL
- o WHAT IS RISK TO SEVERE CORE DAMAGE AND RESULTANT IMPACT ON PUBLIC SAFETY DUE TO LOSS OF AC POWER?
- o HOW DOES RISK COMPARE TO RISK FROM OTHER SCENARIOS?
- o WHAT IS COST OF REDUCING RISK?

DECISION

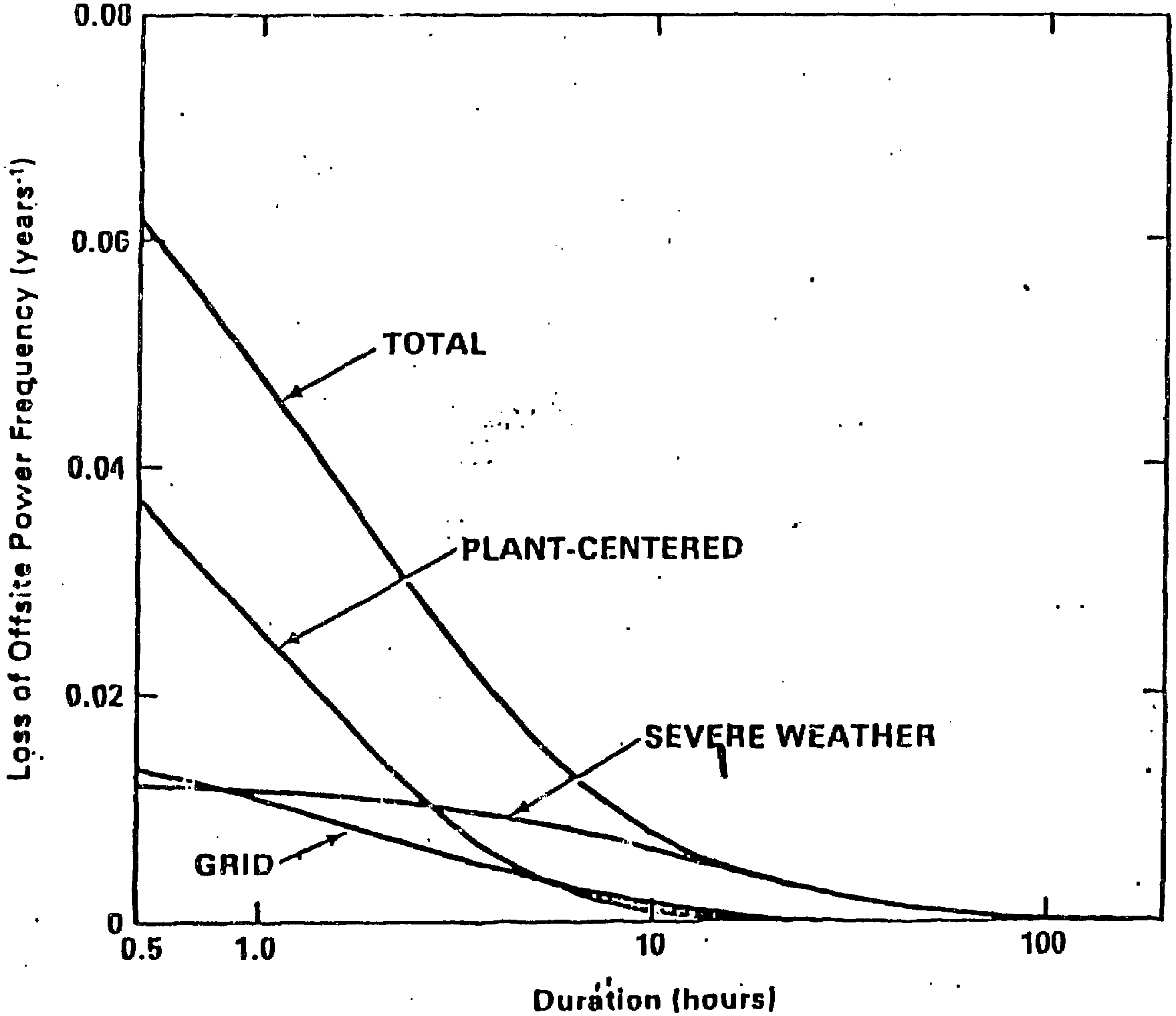
- NOT YET FINALLY DECIDED

  - SUSCEPTIBILITY OF LOSS OF AC POWER IS A PLANT-SPECIFIC ISSUE.

  - DEPENDING ON SUSCEPTIBILITY OF LOSS OF GRID AND DIESEL GENERATOR RELIABILITY, EACH PLANT SHOULD BE ABLE TO TOLERATE STATION BLACKOUT FOR A PERIOD OF TIME RANGING UP TO ABOUT 4 HOURS.

  - PLANTS SHOULD BE ABLE TO DEMONSTRATE, THROUGH ANALYSIS, THE LENGTH OF TIME THEY CAN COPE WITH A LOSS OF AC POWER.

Historical Loss of Offsite Power Frequency and Duration at Nuclear Power Plants



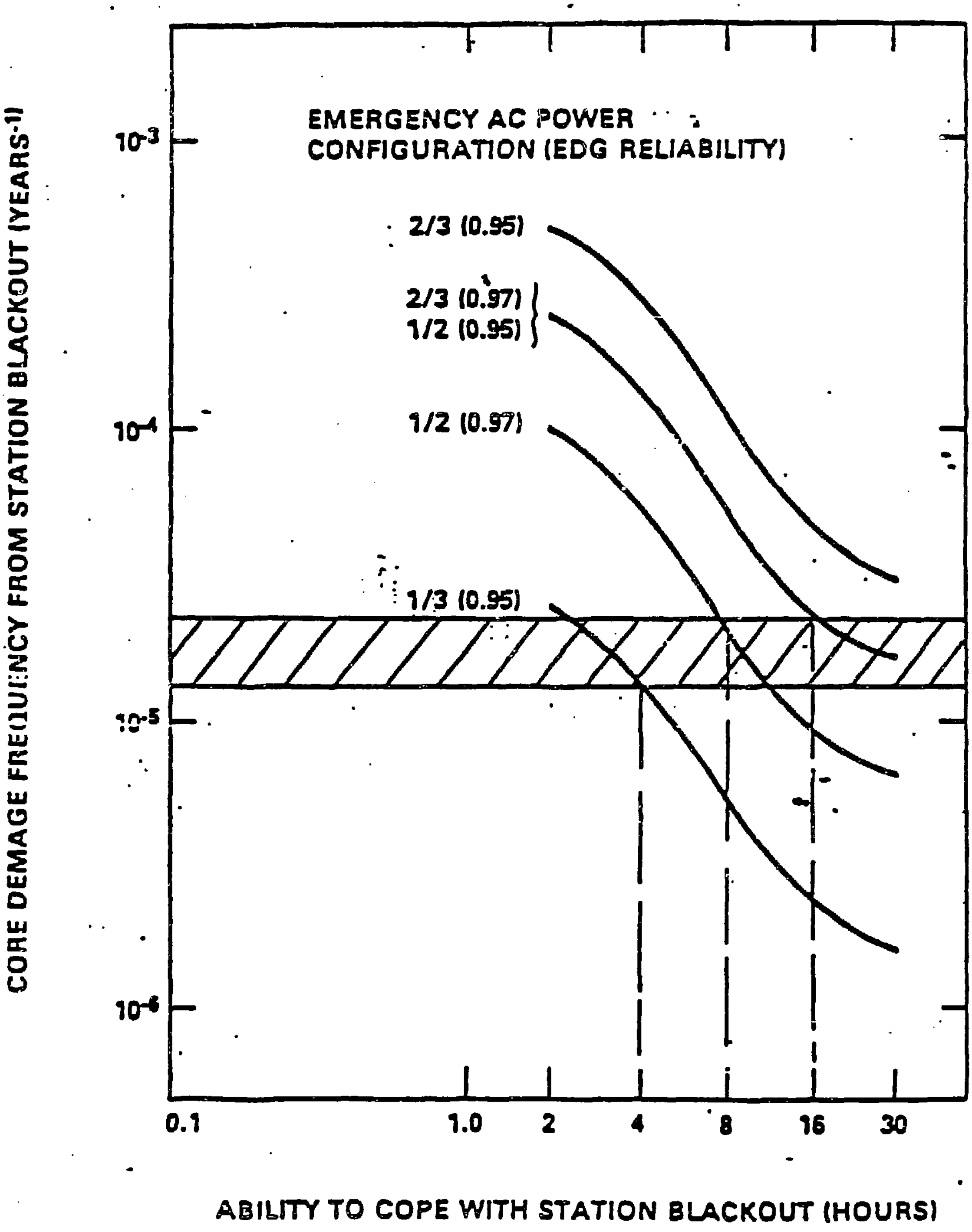
How often is off-site power lost at that plant, and why is it lost? Is it lost because of bad weather? Is it lost for a long time or fairly short? Are the decay-removal systems dependent on AC or only DC? DC power failures. The pump seals on the primary pumps - how will they behave for an extended period of time.

Let's take some of these questions and start to show you graphically on this chart what they really mean. What should you be thinking about? Well, if you compile the data I was talking about, this figure shows that the longer the duration of loss of power, the less likely it is to occur. That makes sense. If you lose power off-site, the utilities usually can do things fairly quickly to restore power. The longer you have to do it, the more things they can do; hence, it makes sense logically. These curves are based on actual experience and data in the United States. So that's the first ingredient, how long will power be lost and its likelihood.

Next is the ability to cope with station blackout. Here we have the core damage frequency on the vertical axis of the chart for a variety of diesel-generator configurations. Again, it makes sense. If you only need one diesel generator and you have three diesel generators, then the damage to the core is obviously going to be at a lower frequency than if you need one of two. That you can see by picking one of three, at .95, compared to two of three. Some of these core-damage frequencies can get fairly high, 10^{-3}, 10^{-4}, for a very poor diesel generator or generators and a poor configuration. So you not only have a question of safety but you also have a question of economics. Even if you don't have a core meltdown with significant damage off-site, if I were in the industry I would be asking myself, "Wait a minute. If I can get these kinds of frequencies for core damage and losing a two billion dollar investment, then maybe I ought to think about adding a diesel generator or improving maintenance." The utility industry is starting to ask these kinds of questions of themselves, based on PRA. How ought they to change their plant because of the economic investment they have in the plant?

This last slide is a very busy one but I wanted to show you the sensitivity of station blackout risk to different containment concepts. I'm not giving you an answer to the question. I only want to make you aware of

ESTIMATED CORE DAMAGE FREQUENCY FROM STATION BLACKOUT EVENTS VS. ABILITY TO COPE WITH STATION BLACKOUT (OFFSITE POWER CATEGORY II)



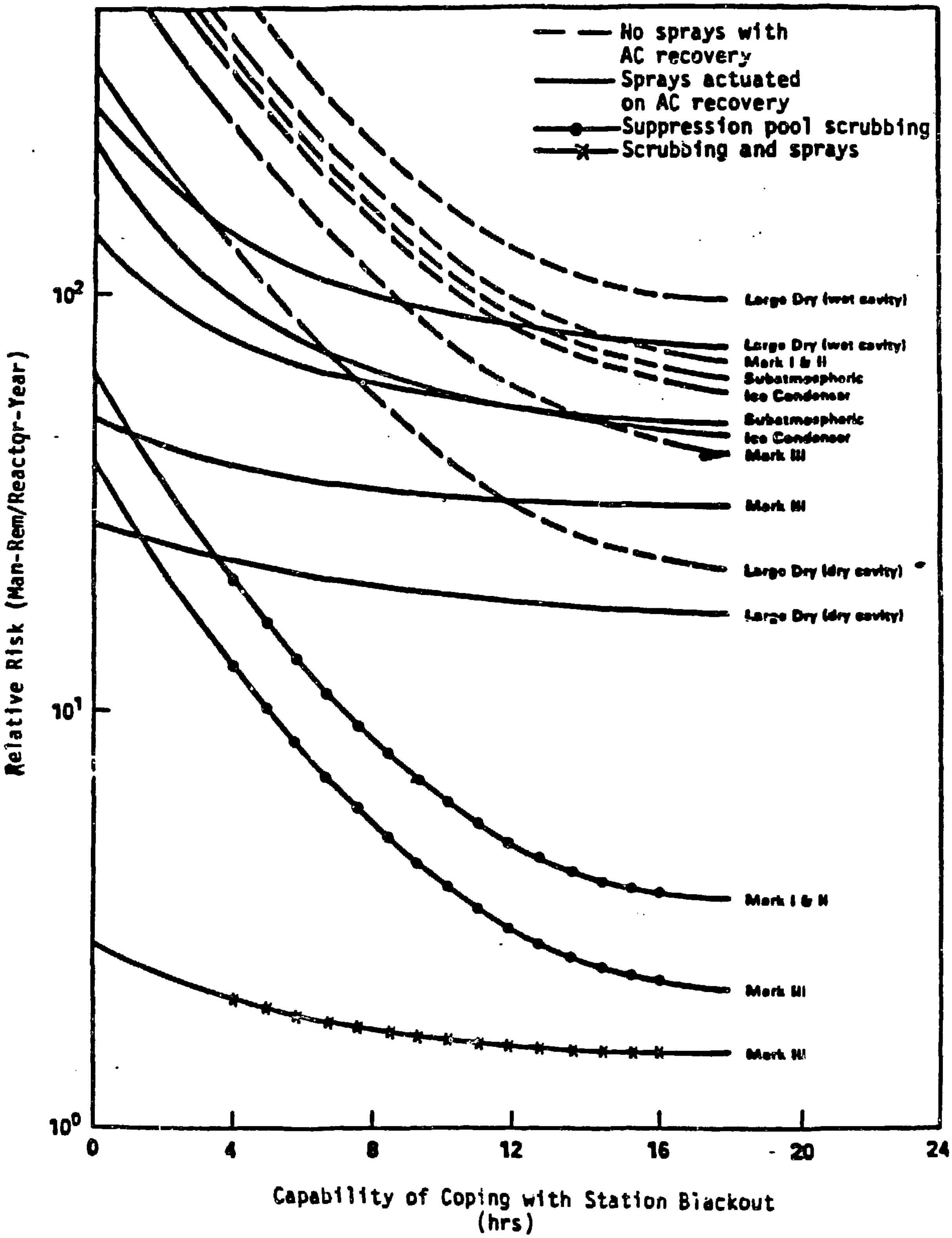


Figure C.1 Station blackout risk perspective for different containments the kinds of information that you want to be asking for. This isn't very sophisticated. This isn't a big, elaborate PRA that takes $10,000,000 to do. This could be done fairly cheaply, fairly inexpensively, and it gives you very substantial insight into a power plant.

The next issue that I want to talk about is sumps. We spent a great deal of time, energy, and money studying the problem. As you know, if you look at a typical power plant, you have a lot of "stuff" inside the containment. Insulation on pipes, for example, and other things that can fall down and collect in the sump. Hence the potential for blocking a sump. If a very large pipe in a PWR at 2200 pounds pressure should rupture you can visualize how much insulation and debris it could create inside that containment. It collects on the bottom. Question. Should we issue requirements that get rid of certain kinds of insulation such as asbestos, the kind of stuff that can break up? Should we require insulation which is made of metal which is not very mobile and can't clog up sumps? If I put the mirror insulation on, that's what this metal insulation is, I can do my inspections a lot easier. I can save man-rem exposure, a very important consideration. I can reduce the amount of radiation people have, so how much is that worth? Eventually, however, you want the real question answered first and that's, "Do I really need to worry about the sump? And whether it's blocked or not in a plant?" How do you get an answer to that? Well, we looked at the results of 15 PRAs and we ranked systems in them according to their relative importance in dominant accident sequences. If you look on this chart, you will see that the sump is <u>least</u> important of all that were analyzed. And as you might expect, the auxiliary feedwater system is the most important. That means, if you are really going to improve safety you want to be doing something with all of <u>these</u> things before you do much with a sump. Now why is that? You start to ask yourself, why did these results come out that way? It gives you some very interesting insights. What's really the only mechanism that you have available to create a lot of debris that can block up a sump? Think about it. How can you really block up a sump? What do you have to have happen? You have to have a very large pipe with a very large rupture to create the dynamic effects necessary to create that kind of debris.

ITEM: A-43 CONTAINMENT EMERGENCY SUMP PERFORMANCE

ISSUE

-- DURING RECIRCULATION MODE AFTER A LOCA, WILL UNACCEPTABLE SUMP BLOCKAGE OCCUR DUE TO ACCIDENT GENERATED DEBRIS, PRIMARILY RCS PIPE INSULATION.

- o PUMP PERFORMANCE
  - NPSH (FLOW REDUCED BY PARTIAL BLOCKAGE)
  - MECHANICAL DAMAGE DUE TO DEBRIS INGESTION
- o SUMP PERFORMANCE
  - AIR INGESTION DUE TO VORTEXING OR OTHER HYDRAULIC EFFECTS
  - SCREEN COLLAPSE DUE TO PRESSURE DIFFERENTIAL
- o IMPACT ON DECAY HEAT REMOVAL CAPABILITY

CONSIDERATION IN DECISIONMAKING

- WHAT TYPES OF INSULATION USED IN PLANTS?
- WHAT TYPE OF ACCIDENT INDUCED IMPINGMENT IS NECESSARY TO KNOCK INSULATION OFF COMPONENTS?
- WHAT WILL BE SIZE OF INSULATION PIECES?
- WHAT IS PATH OF INSULATION TO SUMP?
- ARE FLOW VELOCITIES TO SUMP SUFFICIENT TO TRANSPORT INSULATION?
- FLOW AREAS OF SUMP SCREEN.
- HOW MUCH AND WHAT CONFIGURATION OF INSULATION MUST ARRIVE AT SUMP TO CAUSE A BLOCKAGE PROBLEM?

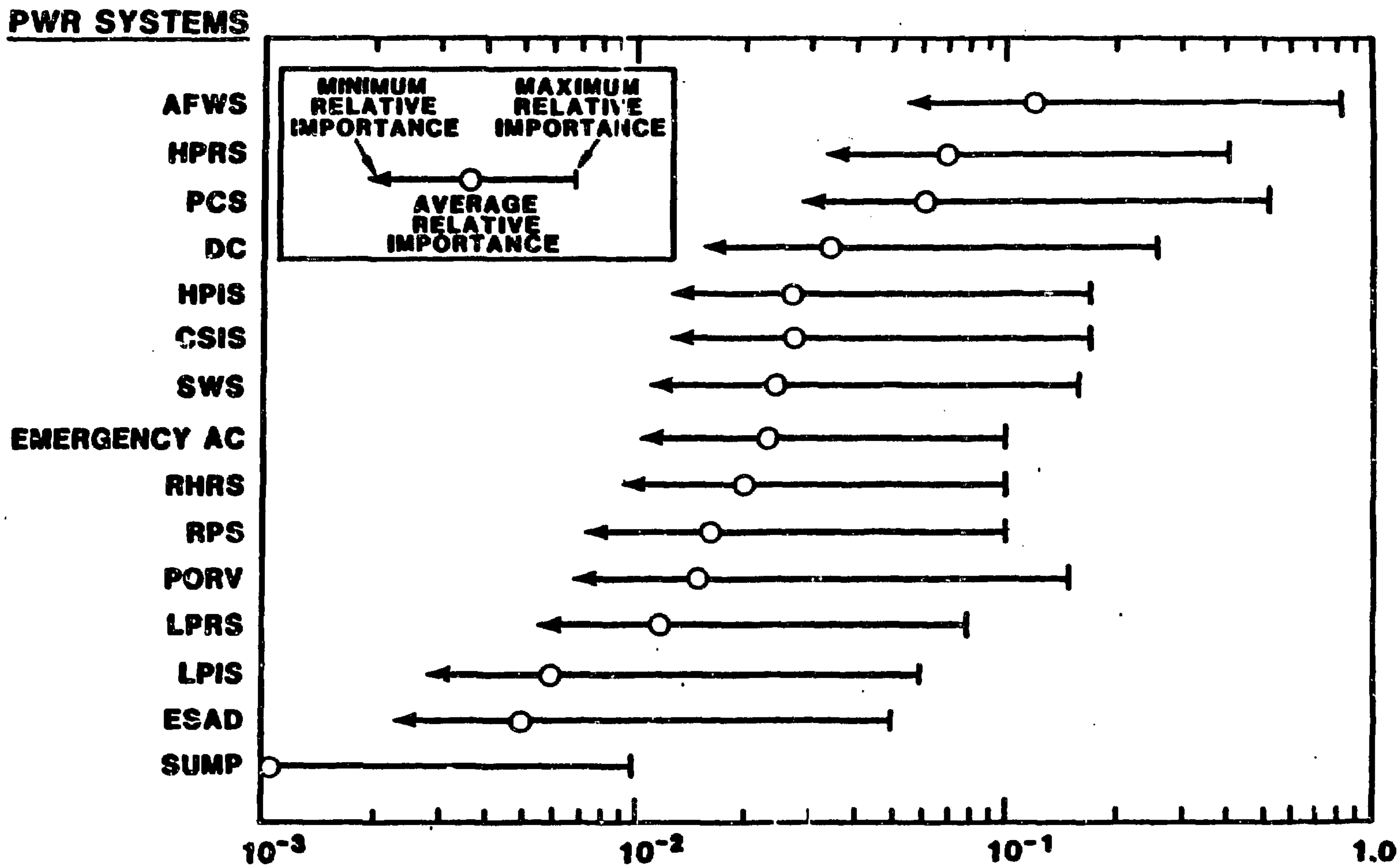
CONSIDERATION IN DECISIONMAKING (CONTINUED)

- CAN INSULATION PIECES THAT PASS THRU INTACT SUMP SCREEN CAUSE PUMP PERFORMANCE/ COOLING PROBLEMS?
- WHAT IS PROBABILITY OF SUMP BLOCKAGE?
- AT WHAT TIME AFTER ACCIDENT WOULD BLOCKAGE OCCUR, IF IT OCCURS?
- SHOULD THIS BE ANALYZED ON A GENERIC BASIS OR PLANT-SPECIFIC BASIS?
- POTENTIAL MAN-REM SAVINGS IF FIX NECESSARY AND IMPLEMENTED.
- COST OF ANALYSIS TO DETERMINE IF FIX NECESSARY AND COST OF FIX.

DECISION (BASED ON STAFF WORK SEEN TO DATE BY CRGR)

- BASED ON NRC GENERIC ANALYSIS USING REPRESENTATIVE PLANTS, IN THE EVENT OF AN ACCIDENT, DEBRIS GENERATED BY THE ACCIDENT WHICH COULD REACH THE CONTAINMENT SUMP WOULD BE INSUFFICIENT TO CAUSE A SUMP BLOCKAGE PROBLEM FOR WELL DESIGNED AND REDUNDANT SUMPS. SUCH DEBRIS WHICH PASSES THROUGH THE SUMP SCREENS WOULD NOT CAUSE A PUMP PERFORMANCE OR DECAY HEAT REMOVAL PROBLEM.

- NO PLANT-SPECIFIC ANALYSIS OR REMEDIAL ACTION IS NECESSARY.



Relative Importance of PWR Systems Considering Dominant Accident Sequences from 15 PRAs

Question. What's the probability of getting that kind of a massive failure in a pipe? Answer. Very, very low. Hence, the only mechanism available to get a situation in which you could be faced with that kind of a problem, the PRA tells you it's not important. It's obvious. The only way you can really get that kind of damage is to have a very, very low-frequency event.

There is another issue, while I have that slide on, that I want to talk about, and that's the issue of a pilot-operated relief valve (PORV). Some plants in the United States do not have them. How do you go about deciding, should I add a PORV? Will I make the plant better? Do I need to make it better? Will it be safer? Well, the first thing that I was interested in seeing is the slide in front of you. How important is a PORV in terms of relative risk? And you can see, relative to the most important ones, a full order of magnitude less. Maybe we really don't need to worry about it. You can actually go ahead and do a PRA, do a probabilistic risk. You don't even need the end result. You don't need off-site consequence. You don't need the consequence end of it. All you need is a level 1 PRA to really get some insights.

With that, I want to put on the last slide. It shows the results of two analyses, one of them is the CE Owners Group (CEOG) analysis, that is, the utilities with Combustion Engineering facilities got together as an owners group and analyzed the plant to try to decide "Should I add a PORV?" Some people think it's obvious. Let's just go ahead and add it. Well, they did an analysis, and what do you think their analysis showed on their plant? If you did the analysis <u>their</u> way, it shows that you would make the plant <u>less</u> safe if you add a PORV to it. Interesting. Make it less safe. Well, the NRC sponsored an analysis; that's the one on the right-hand side. We did an analysis and we played with some of the assumptions. We said, "What's the failure rate of PORVs?" We got the historical data but we didn't stop there. We know how you can make PORVs better than that. Let's assume that we improve the PORV. Well, if you improve the PORV, then it won't fail as often. It'll fail less often, and if you plug that failure-rate that we came up with, what do you get? You get something that says it makes it <u>more</u> safe. Why is there a difference? The real

ITEM: PORVs IN CE PLANTS

ISSUE

- IS A PORV NECESSARY IN CE PLANTS TO REDUCE CORE MELT FREQUENCIES, ESPECIALLY IN LIGHT OF ATWS CONSIDERATIONS?

- IF A PORV IS NECESSARY, SHOULD IT BE AUTOMATICALLY ACTUATED OR MANUALLY ACTUATED?

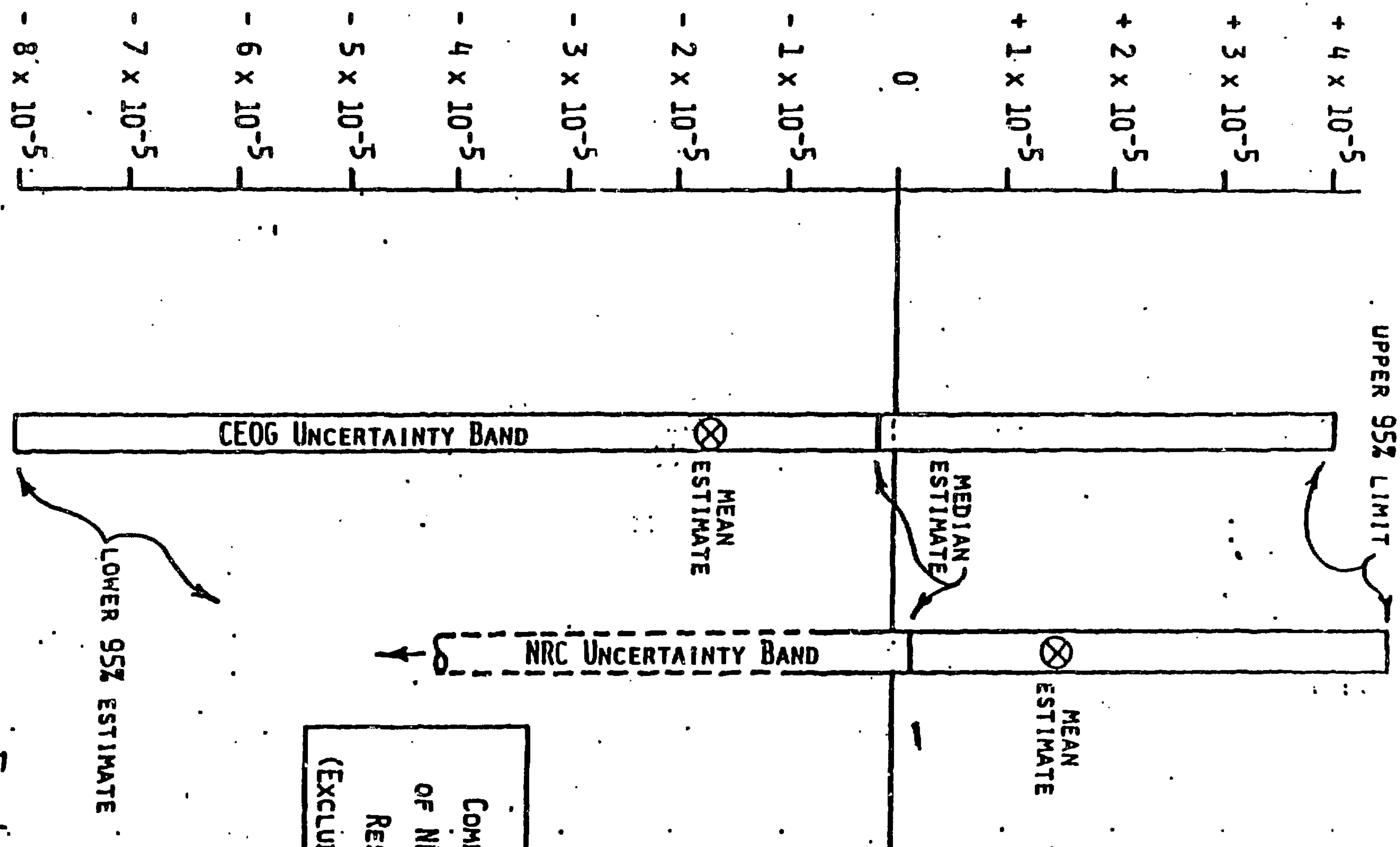
CONSIDERATION IN DECISIONMAKING

- o IMPACT OF PORV ON LOCA FREQUENCY
- o IMPACT OF PORV ON CORE MELT FREQUENCY - AUTOMATIC VS. MANUAL ACTUATION
- o RELIABILITY OF PORV BLOCK VALVE
- o IMPACT OF ATWS FIX ON NEED FOR PORV
- o IMPACT OF RESOLUTION OF A-44 (STATION BLACKOUT) ON NEED FOR PORV
- o FEED AND BLEED IMPACT ON NEED FOR PORV
- o COST OF PORV INSTALLATION

DECISION

- DEFER RESOLUTION OF CE PORV ISSUE UNTIL RESOLUTION OF A-44 (STATION BLACKOUT). AT PRESENT TIME COST BENEFIT OF MODIFICATION IS MARGINAL.

NET REDUCTION IN CORE DAMAGE FREQUENCY DUE TO PORVs (Auto)



Comparison of NRC/CEOG Results (Excluding ATWS)

Enclosure 6

reason there's a difference is that a PORV does two things to you. Remember TMI. The PORV failed open; it stuck; it failed. And hence it created that small loss-of-coolant accident that created the problem they had. It was in fact the starting of it. If it doesn't fail very often, and then you have a need for it, it'll come open as it should and then it'll reclose. It's the potential for the failure open that creates the unsafe end of this. So you want to ask a lot of questions when you're using these kinds of analyses to make sure that you're doing it correctly. But I'm all in favor of PRA. I think PRA is not only good for the regulator, I think it's very, very good for the industry as well. I think I've seen in the last few years a real change in opinions in the industry's view too.

VIII REGULATORY CONSIDERATIONS OF THE RISK OF NUCLEAR POWER PLANTS

Robert M. Bernero
Director, Division of Systems Integration
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission

REGULATORY CONSIDERATIONS OF THE RISK OF NUCLEAR POWER PLANTS

Robert M. Bernero
Director, Division of Systems Integration
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission

I find it very fruitful in this situation to use my own career in the regulatory agency as an illustration of the regulatory considerations I will explain to you. When I first started in the Atomic Energy Commission regulatory staff, I was in reactor licensing, and I worked on the safety review of many power plants, using the techniques available at the time -- the traditional safety analysis. However, because earlier in my career I had worked in Naval reactor programs and space nuclear programs for the United States, I had learned many years ago some of the first principles of probabilistic risk analysis, and in fact used fault-free analysis many years ago in the space program. As the years went by, I went in the Nuclear Regulatory Commission staff, into positions in the standards development organization and in research. Some of you have encountered me in years past in the Office of Research, where I directed the development of probabilistic risk analysis techniques. I have gone back to reactor licensing, where I now direct the systems safety review of all U.S. reactors. We are restructuring our organization, and soon I will direct the licensing and safety review of all of one class of reactors, boiling water reactors. That is a very good symbol of how probabilistic risk analysis has become embedded in our regulatory process. It is a visible and useful part of our process, and I think we use it with care. To emphasize the care, I would like to quote a synthetic proverb from China. Confucius, 2500 years ago, according to the story I like to tell, warned succeeding generations that in PRA the only significant figure is in the exponent. In risk analysis you calculate the probability of something -- a core melt, a severe core-damage accident, an early fatality, a cancer fatality, a genetic defect -- you calculate something. You calculate a probability. But the least useful and most dangerous part is the number itself. You should not treat that number as a precise value. You should not treat that value as something

REGULATORY CONSIDERATIONS

- o PROBABILISTIC RISK ANALYSIS
  - SYSTEM RELIABILITY ANALYSIS
  - SEVERE ACCIDENT ANALYSIS (RELEASES, SOURCE TERMS)
  - CONSEQUENCE ANALYSIS
- o SAFETY GOALS

to test yes or no. It should not make decisions for you. The analysis itself, if you understand it and understand its weakness and strength, can be very valuable.

What I intend to do today is to go through probabilistic risk analysis in its principal elements, as you see on one of the handouts I gave you. And I will treat each of the three sections of probabilistic analysis, then I will discuss safety goals, because when you are measuring or describing risk there must be at least in the back of your mind the question, "What is acceptable?" or "What is not acceptable?" So I will cover first the elements of the probabilistic analysis and then a discussion of the safety goals. Having done that, having covered then what risk can tell us about the regulation of safety in nuclear power, I will then go to the issues that are broader. How, then, have we learned in the United States to regulate nuclear power plants.

I have passed out some color prints of later figures I will use. This one here is a photograph I selected at the last minute, and it is chosen to illustrate, when we are talking about a nuclear power plant, what we are discussing. This illustration is of the Perry Nuclear Power Station in Ohio. It's near Cleveland, Ohio, in the United States. That's Lake Erie in the background. Of course, only in the United States would we put a cooling tower in a nuclear power plant on the shore of a Great Lake. But we do this, in more than one location. The nuclear power plant, after all, is a factory. It produces electricity. There are very many people working in that factory, and consequently it's a very large array of structures. This particular plant has two reactors, Unit 1 and Unit 2. They are boiling water reactors with Mark III containment. The BWR-6 model, the advanced model. And they each have a large cooling tower and, of course, the switchyard to deliver the product, electricity, to the customers. If you look into a nuclear power plant, it is a very complex machine. And for the utility or the company or the government that operates that machine, its primary purpose is to serve the factory function, the turbine generator to deliver electricity. As a safety regulator, there is a fundamental principle that you can adopt. Our experience, after reviewing more than 100 reactors and licensing almost 100 of them by now to operate, is that, if you look in here at the reactor core, that is where the fission products are, that is where safety is protected. If you protect the core, you protect the public. You must observe and regulate low-level waste, radiation, direct radiation, streaming, radioactive liquids, drains, collections -- those are important. That's good housekeeping. You should take care. You should not allow contamination to spread. But when you look at the central safety of the public and for that matter of the hundreds or even thousands of people that might work on that site, you should concentrate your attention on the preservation of the core. So that's why the risk analysis is typically focused on core melt -- the probability of core melt. In the analysis that is done today for PRAs, when we say core melt, the probability or frequency of core melt is not what we calculate. Most analysts calculate the probability or frequency of reaching conditions where the core will be severely damaged, because it is not sufficiently cooled. There are reasons, analytical reasons, why that is done. It is important because it is not the public safety threshold. It is not the line of actual core melt. It is the line of severe core damage which may progress to core melt, which is a convenient thing for the owner of the plant as well as the regulator, because it is at that point, the point of severe core damage, where the threat of core melt becomes more real and the threat of severe economic damage to the owner of the power plant is real. The Three Mile Island plant suffered a partial core melt. It suffered severe core damage, and when you suffer that kind of damage, as in Three Mile Island (there was essentially no injury to the public off-site), there will be grievous injury to the utility that owns the plant. Billions of dollars can be spent cleaning that up, and yet no one was hurt, so there is a congruent interest. If the owner of the plant protects the core, he automatically protects the public safety; and that's something as regulators you should keep in your minds.

Now, let me go to the elements of PRA. You can break probabilistic risk analysis up into three basic activities in which you do things that lead to a useful product. I will touch on some of the names for these

ELEMENTS OF PRA

- o SYSTEM RELIABILITY ANALYSIS
  - ANALYZES INITIAL CHALLENGES AND SYSTEM RESPONSES OF FAILURES
  - RESULT: CORE MELT PROBABILITY
- o CONTAINMENT EVENTS ANALYSIS
  - ANALYZES CORE MELT PHENOMENA AND CONTAINMENT RESPONSE
  - RESULT: ACCIDENT RELEASES (SOURCE TERMS)
- o CONSEQUENCE ANALYSIS
  - ANALYZES WEATHER, POPULATION, AND EXPOSURE PATHWAYS
  - RESULTS: HEALTH EFFECTS, PROPERTY LOSS

products. The first thing you see is this particular element here -- System Reliability Analysis -- here you are looking at the reactor plant, the entire facility, and you are analyzing the challenges to the system: thunder storms, lightning, excessive load, malfunction in the switchyard, in the turbine, in the feedwater pumps, or anywhere. You are looking at challenges, events that require the reactor to do something -- to change its power level, to adjust the feedwater flow, or to trip, to scram. Those challenges should include accidents, maybe a pipe break. Those challenges must be systematically listed, and for each one you must analyze what systems need to respond, what parts of the plant need to react. The analyst will then analyze for that particular plant and its systems how likely is it that they will successfully respond. How many pumps, how many different ways to respond to that challenge exist in that plant? What is the quality, the reliability, of those systems? How must the operator react if they don't work? It's one thing if an operator in a control room is looking at the control panel and, seeing that a certain pump has tripped off, can restore that pump by pushing an acknowledge button and turning a switch, right at the control panel. That's a relatively reliable second attempt. But on some pumps he may have to look, turn to an auxiliary operator, and say, "George, go down in the auxiliary building to that pump and reset the governor so that I may restart it." You see, it's much more difficult. It takes longer. George may not find the pump, or may not have the key to the padlock on the door. So that reliability analysis must consider those things, and you see right away that it's not going to be precise. Who in this room would dare to say the exact probability that George will find the pump, or that George will have the right key for the door? You don't know that exactly, but you know that could be a problem, and you have a systematic way to collect those problems, those questions, and put them in a reliability analysis. So the product, when you finish your work, the product is what we all call core-melt probability. It is actually the frequency, the probability per year, that you will reach unacceptable plant conditions. Those unacceptable plant conditions are those that are most likely going to lead to severe core damage.

The second part of a risk analysis involves a different set of expertise -- containment events analysis. We use that strange English phrase meaning what goes on in the containment (and that means inside the reactor as well) if the core melts. We are no longer talking about low temperature. We are talking about 3000° Centigrade and above. We are talking about molten ceramic material, uranium oxide, molten metal. We're talking about zirconium metal at a temperature far higher than the temperature needed to burn, to react with any oxygen it can find. We are in a high-temperature regime where it is very difficult to get hard data. I think you will hear in Idaho that we have been successful in getting a lot of data. Certainly in the last ten years the amount of information, scientific information, available for containment-events analysis has increased dramatically. The analysts, nonetheless, are experts, the people I like to call the physical chemists, who analyze how the core melts and what it does to the materials around it, what pressure, what gases are imposed on the containment. And then you need the help of structural engineers to say, "If you have those temperatures and those gases and those pressures, when will the containment fail, or will it fail?" And when you complete that analysis, the result tells you what gets out. What radioactive materials get out, when do they get out, and how do they get out of a nuclear reactor which had a severe accident? Here in the United States and elsewhere we use the words "source terms" for that. The information is used by the person who does the next analysis. The person who looks at dispersion wants to know what is the radioactive source, what is the source term that I will use as the basis of my calculation? So the containment and events analysis produce the source terms, while another kind of analysis looks at the reactor site, at the weather conditions around the reactor site, and what the people might do, how far away they live, are they prepared to move in the event of an emergency, and then analyzes what radiation exposure those people might suffer, and calculates from biological data what the health effects would be. The health effects could be measured by how many people die immediately from very high levels of radiation, what is the probability that others might die of cancer later. One could use radiation injuries. We use that term frequently. Radiation injury, in our terminology, means clinically detectable effects of radiation. That is, chromosome changes, blood changes, even hair fall-out, and so forth. We usually use 50 rem whole body exposure as the threshold, the beginning, of clinical effects. There is a probability of early fatalities above 200 rem whole body. Latent fatalities, of course, are proportional to the degree of radiation exposure. We do not use a linear hypothesis curve in the United States. We use something very close to the Beir-III model, in which very low radiation levels are not proportionally effective in causing cancer. So, the results of the consequence analysis will give you health effects.

There is another part that you, as regulators, should consider in your political system. It is a controversial one in the U.S. political system, and that is property loss. Suppose a severe release is expected to take place in the next six hours, you have ample time to warn everyone close to the plant, they all move away and no one suffers excessive radiation exposure. Just imagine that. It's quite easy to imagine with current analysis. If you have that situation and have that release, what you now have suffered is the deposit of radioactive material on the soil of your country, on some buildings, some houses, some factories, some stores, bridges, roads. And that has to be cleaned up. You can estimate with these models quite accurately, I think, or reasonably, how much contamination will be there and where it will reach. What you are unable to estimate is the political and administrative cost to form groups of workers and to take care of the people who live there and to settle on the level of decontamination needed to let people go back to their homes. That part is hard to predict. So this is a very significant question. After the immediate public health threat is taken care of, what do you do about the public property? In the U.S. today there is a great deal of question about whether our agency, the federal regulatory agency for nuclear safety, is actually supposed to regulate public property damage. Is that a legitimate basis for a regulatory change, a design improvement of a reactor? It's a very interesting and controversial question. So, in each country's political system, that is a conscious choice that the government needs to make. Does the agency regulate the

RISK ANALYSIS
A LOGICAL BASIS FOR REGULATION

- o ACCIDENT PREVENTION
  - HIGH DESIGN STANDARDS
  - EMERGENCY COOLING SYSTEM
- o ACCIDENT MITIGATION
  - CONTAINMENT
  - REMOTE SITING
- o A MEASURE OF SUFFICIENT SAFETY

economics and the risks to the economics, as well as the risk to the safety?

Now, in risk analysis, and notice I leave out the word "probabilistic", at least for awhile, we have a logical basis for regulation. If you turn back and think about it, you can say, "Ah, well risk analysis really gives me a measure of when and how accidents occur, so to prevent accidents I can do certain things. I can have systems like emergency cooling systems, to act if I do have an accident." If I look at the consequences, I can easily see that it is worthwhile to have mitigation, lessening of those consequences, and I can use things like containment, or remote siting. In a country like the United States, we have vast areas of land. We can put nuclear reactors farther away from the population. It's easier for us to do that. It's often a problem for other countries. When they look at our siting criteria, they say, "You know, you are an example. It's very difficult for us in other countries to live with that." Where can you go in some countries to get that far away from people? In risk analysis, you have a reasonable way to know when you are safe enough. For more than 20 years we have been regulating nuclear reactors in this country and in many other parts of the world. In every case, the people have made a judgment that this is sufficient safety, or this is <u>not</u> sufficient safety, and if not, add a pump, or change a procedure, or change a structure, or change a location. So the judgment of sufficient safety is always made. It is not always made with complete numbers, or quantitative measure. Historically, if you go back to the early analysis of reactors, the Brookhaven report WASH-740, is an example. The early analyses were much simpler than what we can do today, for very good reasons. First of all, they did not have an extensive experience in reactor design to base their analyses on. In those days, in the 1950s, we were just building the first reactors. In addition, the availability of computers to do complex analyses was limited. Much of that developed in the last 25 years or so. And as a result, the old analyses tended to be, "I'm not so sure how likely the accident is to happen, but I will try to look at the physical chemistry. I will look at the source terms." And so there tended to be an assumption that the

EARLY ANALYSES

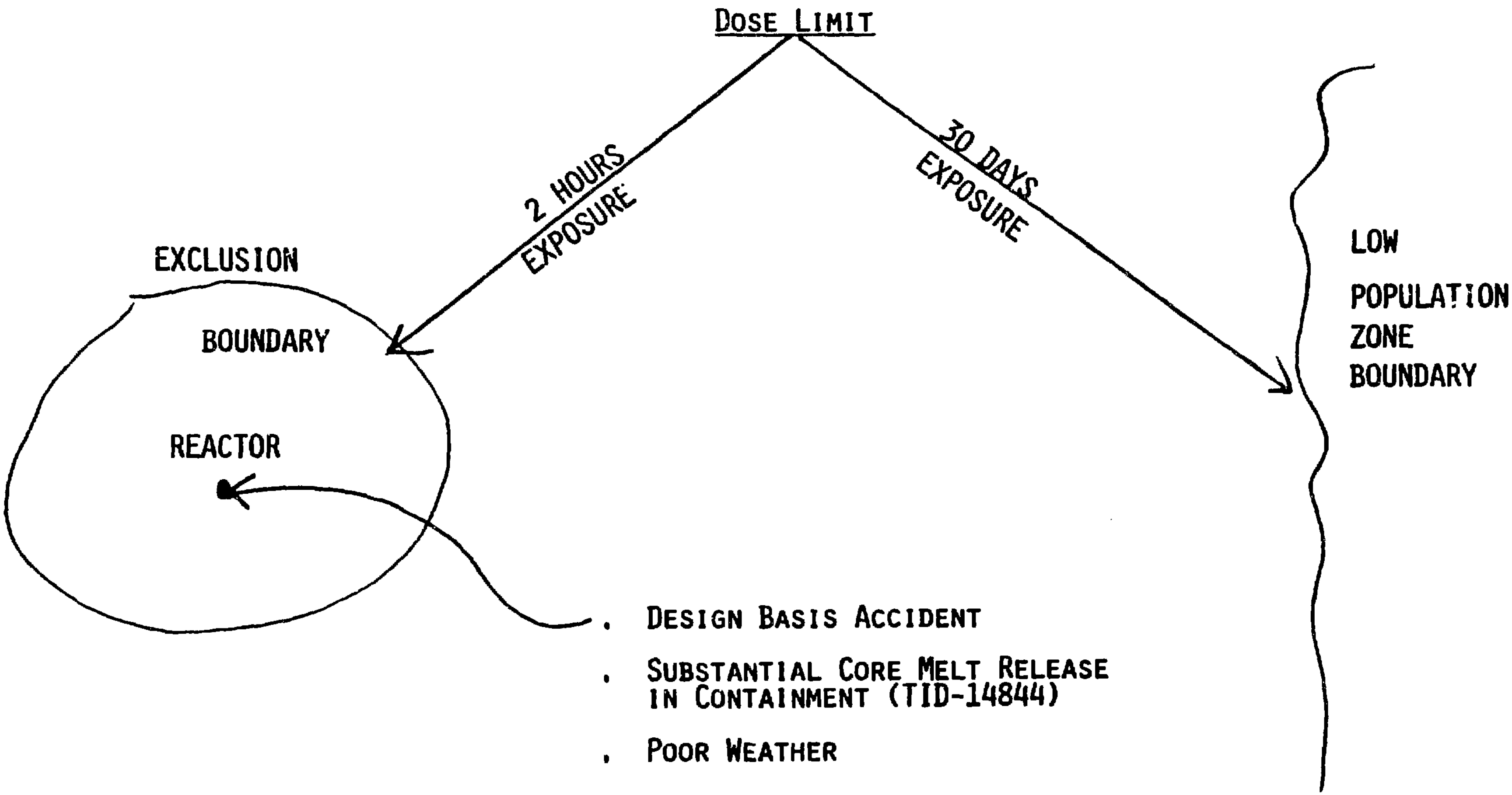
- o LITTLE RELIANCE ON ACCIDENT PREVENTION
  - NO EXTENSIVE EXPERIENCE IN DESIGN
  - NO METHODS FOR COMPLEX SYSTEM ANALYSIS
- o HEAVY RELIANCE ON ACCIDENT MITIGATION
  - BOUNDING ANALYSES OF RELEASES

reactor core would melt, and then a heavy focus on what will get out. That led to the basic decision to have a containment on all reactors, to have a large containment in order to have a safety margin. Those early analyses led to a regulatory approach which, as I look back on it, was not bad. It may have been too cautious, but it led to the production of quite safe machines. First of all, there was the philosophy of defense in depth. And that philosophy was, "Design the machine to very high standards, so that it won't break. Then design the machine so that, if it does break, there are different systems, or redundant systems, to cool it and to prevent the core melt. And finally, just in case it does break, and it does fail with the extra or emergency systems, provide a containment and remote siting and even emergency movement of the people as additional protection of public health and safety." Obviously, you don't want the severe accident, but one defense after the other stands between the occurrence of the accident and public health. So that defense in depth basically went through with these parameters and required substantial protection for the public in each area. That regulatory approach led to our classical siting calculation. I have this illustration in your notes. We had, in those days, no clear way to measure off-site risk. So, in order to get the safety calculation of a severe accident, what we did, and this was more than 20 years ago, was we took the reactor at its site and said, "Let's have the design basis accident." This is the worst pipe break, the largest pipe in the system is assumed to break most abruptly, and it produces the steam pressure in the containment that's the maximum expected for that loss of coolant accident. And then there is a substantial core-melt release assumed. The regulation document merely said that you should postulate or assume that this amount of radioactivity is loose in the containment. Now, one of the weaknesses of that was, it assumed that the radioactive fission products were there, but did not assume the forces, the pressure, temperature, corrosion forces of core melt. And later on, we'll see that that made a difference. But then, the assumption also said, "Assume poor weather. Look at the weather that you find at that site and do this calculation for bad weather." That's very stagnant weather, no mixing, no dispersion, or very little. And then the dose calculation is done for

BASIC REGULATORY APPROACH

- o DEFENSE IN DEPTH
- o PREVENT ACCIDENTS
  - HIGH QUALITY SYSTEMS
  - DESIGN FOR NATURAL EVENTS
- o RESPOND TO ACCIDENTS
  - DESIGN BASIS ACCIDENTS
  - NO SINGLE FAILURE
- o MITIGATE ACCIDENTS
  - CONTAINMENT
  - REMOTE SITING

PART 100 DOSE CALCULATION



- DESIGN BASIS ACCIDENT
- SUBSTANTIAL CORE MELT RELEASE IN CONTAINMENT (TID-14844)
- POOR WEATHER

two hours, at the site boundary, which is where the fence is, basically, where people are kept out. And for 30 days at the boundary of the low-population zone. In most U.S. reactors, that's a distance of about three miles. The purpose of the two calculations, one for two hours and one for thirty days, was to reflect that, if you had this accident you would warn the people, and they would move away, in automobiles, or buses, or even on foot, so that the ones closest to the plant would be exposed for no more than about two hours, and the ones farthest away would be for 30 days. That was a reflection of emergency planning. And it's interesting to note that, having chosen that sort of approach, we actually achieved a very good degree of safety, when we finally learned how to look at it more systematically.

A fresh look, a very historic event in this field of reactor safety, was completed in 1975, and it was the famous reactor safety study. I put the other two names here. It's frequently called the Rasmussen Report or WASH-1400, and it did a probabilistic risk analysis for the first time of two large reactors. Now, by this time we had design and operation experience. We had big reactors in almost mass production, so that we could take a typical commercial reactor and look at it. We could even look at two types, the pressurized water reactor and the boiling water reactor, which are used in the U.S. and in many other places in the world. That study included, not perfectly, but did include all elements of a PRA. It included the probability of core melt, formally calculated, the containment events and releases analysis, and consequence analysis for six different sites in the U.S. Now, the way that it was done was a little bit complicated, but basically they tried to reflect the fact that in the U.S. you can have quite different weather and climate conditions at different reactor sites. I think it's important to appreciate what they did. Each challenge to that reactor, a LOCA (loss of cooling accident), or a steam-line break, or whatever else, a transient of some sort, has to be carefully analyzed for how the feedwater system responds, the main steam isolation valves, the relief valves that go down into the suppression pool, and question after question after question has to be asked and organized. That's the key of it,

A FRESH LOOK - 1975

- REACTOR SAFETY STUDY (RASMUSSEN REPORT, WASH-1400)
- PRA OF TWO LARGE REACTORS
  - SURRY PWR
  - PEACH BOTTOM BWR
- ALL ELEMENTS OF PRA
  - PROBABILITY OF CORE MELT
  - CONTAINMENT EVENTS AND RELEASES
  - CONSEQUENCE ANALYSIS

REVIEW OF WASH-1400

- LEWIS COMMITTEE (NUREG-0400) 1978
  - UNCERTAINTIES UNDERSTATED
  - INCOMPLETE
  - A MAJOR IMPROVEMENT IN SAFETY ANALYSIS
  - CHANGE FOCUS OF REGULATORY ATTENTION

- MAJOR DIFFICULTIES IN ACCEPTING OR USING RESULTS
  - HOW TO INTERPRET UNCERTAINTIES
  - DESIRES TO USE RESULTS FOR PREFERRED PURPOSES (RISK IS LOW ENOUGH TO CURTAIL REGULATION OR UNCERTAINTY IS HIGH SO USE RESULTS ONLY TO INCREASE REGULATION)

- LITTLE USE UNTIL TMI ACCIDENT

to organize all the questions to make some sense out of it. Even things like how much will go through the standby gas-treatment system, or how much will go directly out into the atmosphere? So the system reliability analysis is a very major undertaking, and the results will depend on the level of detail that is plant-specific. One of the great mistakes of the reactor safety study was that we thought that one BWR was close enough to all the other BWRs that, if you analyze one, you have a good sample of the whole population. And the same for the PWR. And what we know now is: Be careful! There are many little differences. You can find differences in the feedwater; you can find differences in off-gas treatment. Many small differences from one plant to the next can have a very significant difference on the results. And therefore, one of the lessons we have learned since the reactor safety study and must not forget -- you can get some approximate understanding from one PRA for another plant, but it is something you must do with care. Look for differences that can be significant. The same is true in analyzing when the core melts. Does it land here? There's concrete here. Does it stay on the concrete, or does it attack the steel shell and perforate the shell? If it stays on the concrete, the material (gases, fission products) can bubble out through the suppression pool. And if you vent here, you have a filtered vent containment system. A boiling water reactor, if it works that way, has an excellent safety potential. In fact, it even has an excellent economic protection potential because the only thing that would come out is the gases, which don't contaminate the land and which are not enough to kill people. So, you have to analyze from one plant to the next how big is the concrete, what does the wall look like, how might the core-melt material distribute. I want to emphasize that the analysis is complex and that the analysis is unique to the plant you study and should only be sent to another plant with care.

Now, if you go back and look at the results of the reactor safety study (I have a rough summary of them here), they came out with some very important information. They said that, for those two types of reactor, the approximate probability of core melt (remember, severe core damage) is 5 10^{-5} per year. That's one chance in 20,000. At that time, there was a good deal of question that the reactor safety study did not account for certain things, so that, if anything, that number would come higher if you learned how to analyze better. The releases that they calculated were high if the containment failed early, but for most cases, they said, "You know, the containment will fail." They assumed that the containment will always fail, but for most cases the containment failed in a way that filtered a lot of the material. As a result, the risk was estimated to be low. The risk was low because the usual outcome of a core melt was not a high level of radiation release. The lesson of the reactor safety study, at least to me, was: "The likelihood of a core-melt accident is far higher than you think it is." So many of us at that time had an expectation that we had so many protections that we were virtually preventing severe accidents. And this was saying, "No, no, the probability of these events is far higher than you think it is." And the other lesson was, "And the consequences of these events are much lower." As a regulator, I am pleased to say, in a way, that what we've learned since then is even stronger in that direction. We have since learned that the probability is higher than WASH-1400 predicted, unless you do something about it. And we have since learned that the consequences are even lower. WASH-1400 changed our attention; not right away, but it changed our attention. It said, "All you people as regulators ever talk about is the worst earthquake, and the worst loss-of-coolant accident, and you postulate these terrible things and design everything for them. You don't pay attention to simple events -- transients, small loss-of-coolant accidents, operator errors, more than one thing failing, two errors, three errors. Not a single failure, but a triple failure, three different things, or common-cause failure." And so, WASH-1400 in 1975 told us, "Turn your attention as regulators to the real causes, the significant causes of severe accidents." It took us four years and one accident to turn our attention,

Now, as many of you know, there was a review of risk analysis. The NRC hired a group called the Lewis Committee, Professor Harold Lewis from the University of California in Santa Barbara, and they published

WASH-1400 RESULTS

- o PROBABILITY OF CORE MELT: ABOUT 5 X 10^{-5}/YR
- o RELEASES: HIGH IF CONTAINMENT FAILS EARLY
- o CONSEQUENCES: CAN BE HIGH, BUT TYPICALLY LOW
- o LESSONS: - LIKELIHOOD HIGHER THAN EXPECTED BUT CONSEQUENCES LOWER THAN EXPECTED
  - LOW RISK
  - TRANSIENTS AND SMALL EVENTS, OPERATOR ERRORS, MULTIPLE FAILURES ARE THE PRINCIPAL CAUSES - NOT BIG EVENTS

the document "NUREG-0400", their committee report in 1978. They said the uncertainties are understated. They said it's incomplete; it left things out. And that was true. But it also said this is a major improvement in safety analysis. It's very useful; use it. Use it carefully, but use it. That changed the focus of regulatory attention from the earthquakes and the large LOCA to transients, operators, and other things. We had a difficulty in this country because it's very difficult to interpret uncertainties. Uncertainties are things that do not lend themselves to precision and precise definition. I have so many times been asked by people, "What confidence level do you have in the core-melt release term or source term? What confidence level do you have in the calculated consequences off site?" I can't answer those questions. Not quantitatively. If you go to the Coca Cola factory where they make bottles or cans, and they make millions of them every year with big quality control programs, you can ask the manager of that plant, "What confidence level do you have that the Coca Cola bottle will stay intact when you fill it?" He can answer that question, because he has an enormous pile of data, and he can give you precise numbers. But when you are analyzing refractory materials melting at temperatures above 3000°K with no such voluminous data base, you're statement of uncertainty is qualitative, not quantitative. What you do is a sensitivity analysis, not an uncertainty analysis. You say, "The evidence appears to show this outcome, from physical laws, from natural phenomena, understanding what is technically possible." How sensitive is this judgment to the possibility I may be wrong in one respect or another. And if I am, does that change the outcome significantly. So you do that. I don't even like to use the words "uncertainty analysis". I prefer to call it "sensitivity analysis", because you're analyzing how sensitive is my regulatory decision to the unknown in my analysis. In the U.S. at that time, and even now, we had a very large body of opinion on both sides of the fence. We had people who said, "We have too much regulation; here, the reactor safety study says 'Risk is low'. Leave us alone. Stop regulating us. We're safe; we have a lower likelihood of hurting someone than a meteor hitting them on the head." Those are the people that said, "Turn your back on it. Stop. Don't pay any attention to safety." And there were others who went to the other extreme and said, "No, there shall be no balance. Everything you know and everything you do should be used for more and more and more regulation." And the underlying reason was they will never be safe enough. Cancel them, close them, get rid of them. And we had this contest, and in the middle of the contest, risk analysis was not used. We had the accident. The accident at Three Mile Island completely corroborated what WASH-1400 said about the sources, the causes, and the nature of accidents, including that the consequences. Even with such severe damage, even partial melting, as we know now, the off-site consequences can be low.

So we did not use it until TMI, but after TMI there was a dramatic change. There was a major renewal of PRA. People recognized what it could do, and we immediately began to use risk analysis in our regulatory process. The first uses and even the present uses are quite careful and worth noting. TMI is a pressurized water reactor. You may recall from reading of the accident sequence that, early in the accident, the auxiliary feedwater system was disabled for eight minutes. As it turns out, that had little effect on the accident itself. If it were not disabled, I don't think it would have changed the outcome, but it got some attention on the auxiliary feedwater system. I'm sorry to say that in those days the auxiliary feedwater system of the pressurized water reactor wasn't even treated as a safety system. I remember, in 1972, when I first joined the Atomic Energy Commission regulatory staff, my very first case was to review a pressurized water reactor. I opened the safety analysis report and looked for the decay heat removal function, and I found the most important part of decay heat removal, which they called "auxiliary" (not "emergency" but auxiliary) feedwater system, and I found it without quality standards, with virtually no redundancy, and I found it in Chapter 10, "Non-Safety Systems Balance of Plant". I went to my supervisors and said, "What is this? This is the most important, the most frequently challenged system in a PWR." I had constructed PWRs, and it wasn't even a safety system. What we did at that time was to take all the existing PWRs and we said, "There are three basic challenges to an auxiliary feedwater system. When you lose main feedwater,

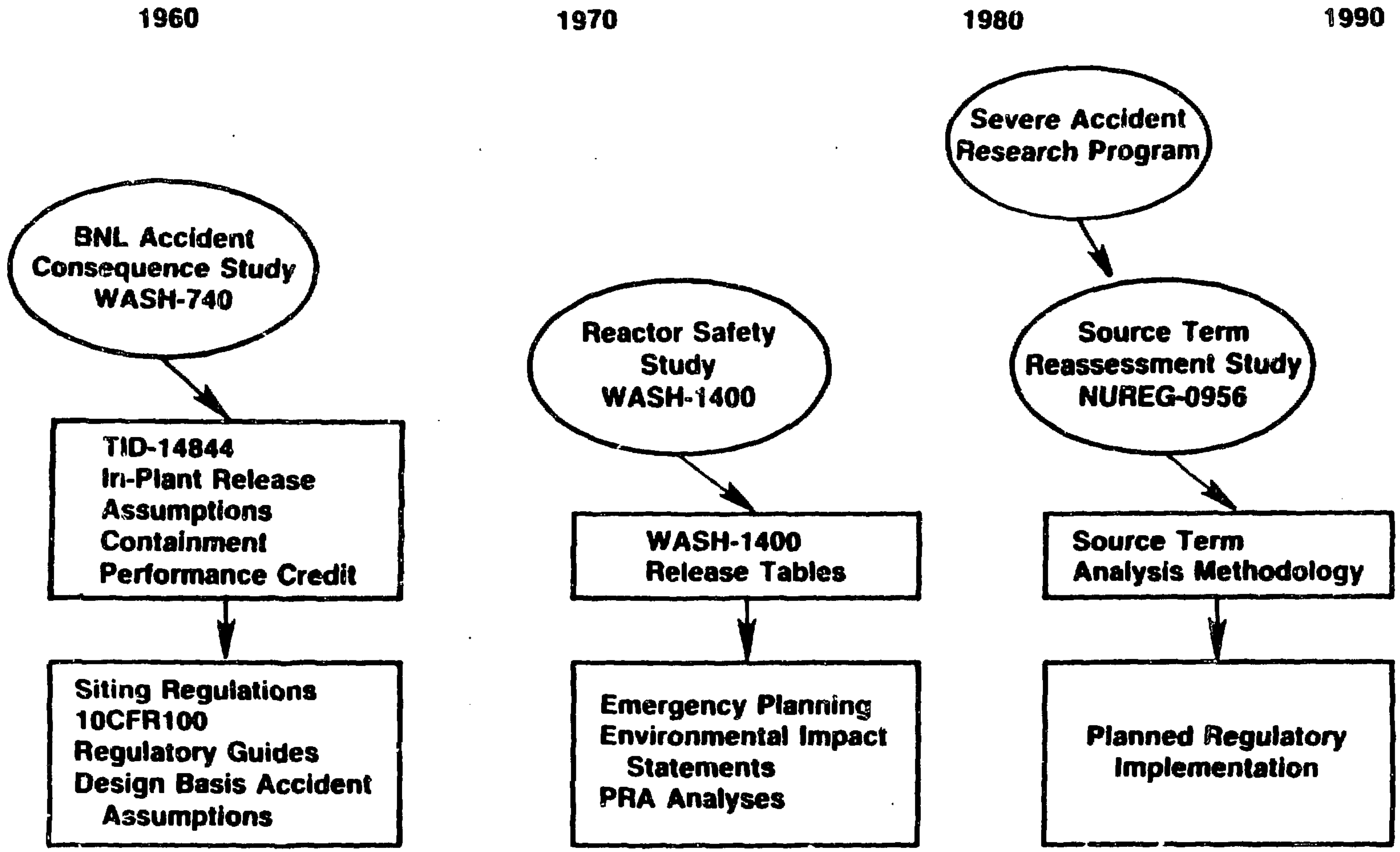
POST TMI

- MAJOR RENEWAL OF PRA USE
- STUDY OF ALL PWR AUXILIARY FEED-WATER SYSTEMS
  - ANALYSIS OF SYSTEM RELIABILITY
  - SPECIFIC CHALLENGES FOR SYSTEM
- MANY PLANT PRA STUDIES
- PRA PROCEDURES GUIDE (NUREG/CR-2300)
- PRA REFERENCE DOCUMENT(NUREG-1050)
- SAFETY GOALS
- SOURCE TERM STUDIES

when you lose main feedwater and also lose off-site power (it's a variation), and last, when you lose main feedwater, off-site power, and on-site power (what we call station blackout)." We did a comparative reliability analysis for those three challenges of that system in every pressurized water reactor. And we found a difference of 100 times in the reliability of one versus another. In other words, some systems were 100 times more reliable than the same system in another plant of the same type, licensed by the same agency for use in this country. That led us to judgments that, for instance, no auxiliary feedwater system shall be without at least one pump train that can operate without AC power. Every one of them must have at least one train capable of independent operation without AC power, steam-driven or diesel-motor-driven, or something. In addition, we require diverse motive power. You should have electric and diesel, or electric and steam, or electric and something else, but not all-electric, or not all-steam, because of common cause failure problems. We went on from there to many other applications of PRA. We've done many plant PRA studies. The NRC, I'm proud to say, has been a leader in this. We've sponsored very many PRAs of plants in order to develop the methods. At the same time the U.S. industry has gone very far. In the U.S. we now have 25 or 30 PRAs. About two years ago, we joined with industry, the scientific community, the governments, etc., and we published the procedures guide on risk analysis, which is a very useful document. It describes in great detail what risk analysis consists of, how it's done, and what it costs, for that matter. Recently the NRC published a document associated with our safety goal work which describes what we have learned so far from the study of risk analysis, quantitative risk analysis. That's a very useful document; I recommend it to you. It is, of course, focused on U.S. experience, but it gives you a good idea of how much we have learned. And then, in the arena of further development, we have safety goal work, and source term studies, which have become of very strong interest here and in other parts of the world. We're focusing on the physical chemistry and the containment, what actually gets out if you have these accidents.

Let me turn briefly to the source terms, because that is so current and so popular a subject. I put this figure in here to give you a historical perspective about source terms, because they are a basis for the regulator to work. If you go back to the early days, WASH-740 didn't directly lead to but was related to the work TID-14844, which was the basis of that dose calculation I showed you. TID-14844 is a very significant part of our regulatory process right now, in many ways, because it was not only the base of that calculation, but it became the design basis source term for filters and for all sorts of other things. Basically, if you look at the lower left-hand corner, there are siting regulations, regulatory guides and design basis accident assumptions, that go right into the regulatory process. In 1975, in the reactor safety study, we used it in only one area, and that was the emergency planning zone, 10-mile planning zone, was derived from a WASH-1400 analysis. We did start using it in environmental impact statements. In 1980, the Nuclear Regulatory Commission said, "Look, in the future, when you speak to the public about a new nuclear plant, give them the best description you have of severe accident risk." So, if you look, we have published dozens of environmental impact statements. These are the reports we make under the National Environmental Policy Act, NEPA. In those, we describe to the public, to the best of our knowledge, how the cooling tower will affect the weather, what it will do to the water, to the wildlife, to commerce, and what the severe accident risk is. We describe this using probabilistic analysis similar to, but now even more advanced than, WASH-1400. Now, this most recent work is going on right now and it is something that you, as regulators, could benefit from. We have just published, from our severe accident research program, this document, NUREG-956. When I say just published, I mean about three weeks ago. It is a methodology. It describes how you would analyze accident releases today as compared to the old methods. It is a much more refined, but unfortunately a much more complicated, method now. It puts us in a position to make a complete reevaluation of all of these old uses of source terms, all of these old TID-14844 uses. So we're in a position now to do this reevaluation. However, the complexity is something that you should not forget. I have just this illustration here.

Source Terms in the Regulatory Process



Progression of Source Term Techniques

TID-14844 Assumptions

Release to the Containment Atmosphere

100% of Noble Gases
25% of Iodine
1% of Solids

WASH-1400 Release Tables

Release to Environment

Category	Xe	I	Te	Ru . . .
PWR 1	1.0	0.7	0.4	0.4
PWR 2	1.0	0.7	0.3	0.02
Etc.				

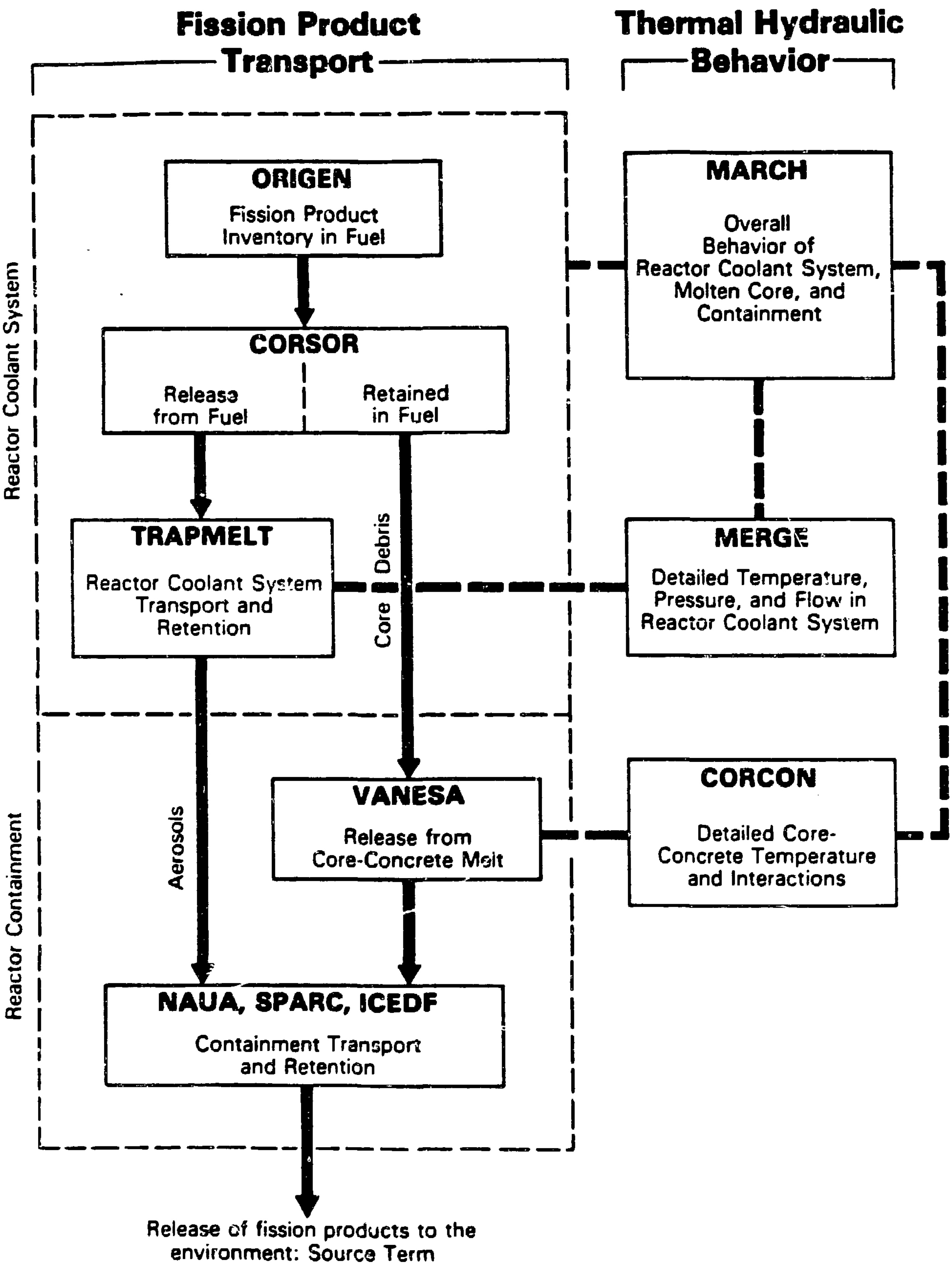
NUREG-0956 Methodology

Analysis Techniques for a Specific Sequence in a Specific Plant

When we say "source terms" in TID-14844, you look at the top column, released to the atmosphere in the containment, that's <u>inside</u> the containment, 100% of noble gases, 25% of iodines, and 1% of solids. So all you need is the size of your reactor. And then you just analyze containment leakage. That's very simple; that's very crude; and that's wrong. It does not represent physical reality. When WASH-1400 came out, it said, "A PWR is a PWR". If I analyze Surrey, I have analyzed all of them. So it broke release categories, and these are release outside of containment, it broke them into categories. You see here "PWR-1, PWR-2," etc. It went through PWR-1 through -9. Had nine releases. The first seven were associated with core melt, and the last two with spent fuel accidents. They said, "For each release category, here is the fraction of noble gas (I just illustrate with xenon), here is the fraction of the iodine, so 70% of the iodine, here is the fraction of the tellurium, the ruthenium, etc. etc. So it not only gave the fraction of each group of fission products, but it gave the time of release and an energy of release, so that you would know does it go high in the air or does it go out as a low plume. But when they did (and it's a very complicated process) the systems analysis and the containment analysis, they looked at the physical parameters and made a decision to put it into bins, or groups. And the groups are PWR-1, PWR-2, PWR-3. So that the source-term table, in WASH-1400, is just a list, PWR-1 to PWR-7 for core melts. And then it gives the time, the energy, and the fraction for each radionuclide. I wish we could do it that simply now. We can't. What we have with our new methodology is a much more complicated thing because we have the curse of wisdom on us. We know too much. We know what's wrong, and we know what's right now.

If you look at this source term code suite, it is a set of analytical codes. Down the left-hand side here, you are analyzing the movement or transport of the fission products, starting with the ORIGEN code, which analyzes how many fission products you have. Then CORSOR computes how much comes out of the core as it heats up. TRAPMELT computes how much is trapped in the reactor coolant system during the release. Farther down you have VANESA. It treats the core melt reaction with the

Source Term Code Suite



concrete, how much is trapped or released there. Lastly, NAVA SPARC or ICEDE, which analyze how much settles in the containment, how much is scrubbed by the suppression pool, and so on. However, you can't address fission product transport unless you simultaneously go down the other column and calculate thermal hydraulic behavior. You get temperature, pressure, natural circulation, etc. from the MARCH code, which calculates core melt. The MERGE code handles the heat transfer inside the reactor coolant. In the old days, WASH-1400, we used MARCH for core melt outside. We don't any more. We use CORCON, core concrete interaction. So these must be done simultaneously. It's very complex, and it means you can't treat all PWRs alike. So it is good to know more than you knew the day before, but it is often a thing that causes you great difficulty.

One of the things we were concerned about in doing this work at the NRC was the scientific integrity of this work. After all, we're acting in the public behalf for safety regulation. We want to be sure that we have scientific integrity. We hired an eminent group. It's the American Physical Society, which has had a long history of public service by scientific comment on items of national interest. I went to them. At that time I was in charge of source term studies. I went to the American Physical Society and said, "I would like to provide the funds, but not a control, so that you could get a blue-ribbon committee, a committee of experts, independent of us, to look at the scientific quality of what we are doing, and tell us what you think. I want no control over the result. I don't want to pick the people. That's for you to do." And they did that. The American Physical Society picked a committee and that committee reviewed our work, the work of our national laboratories. They reviewed some of the foreign work. They reached the conclusions shown on this slide. These are the American Physical Society conclusions. They made the report early this year. They said, "There has been considerable progress since WASH-1400. And in a number of cases, the releases are significantly lower." They point out that containments are really stronger than we thought they were. WASH-1400 assumed the containment would always fail. That's not true. The containment will hold together many, many times. The current studies, with their complexity, include

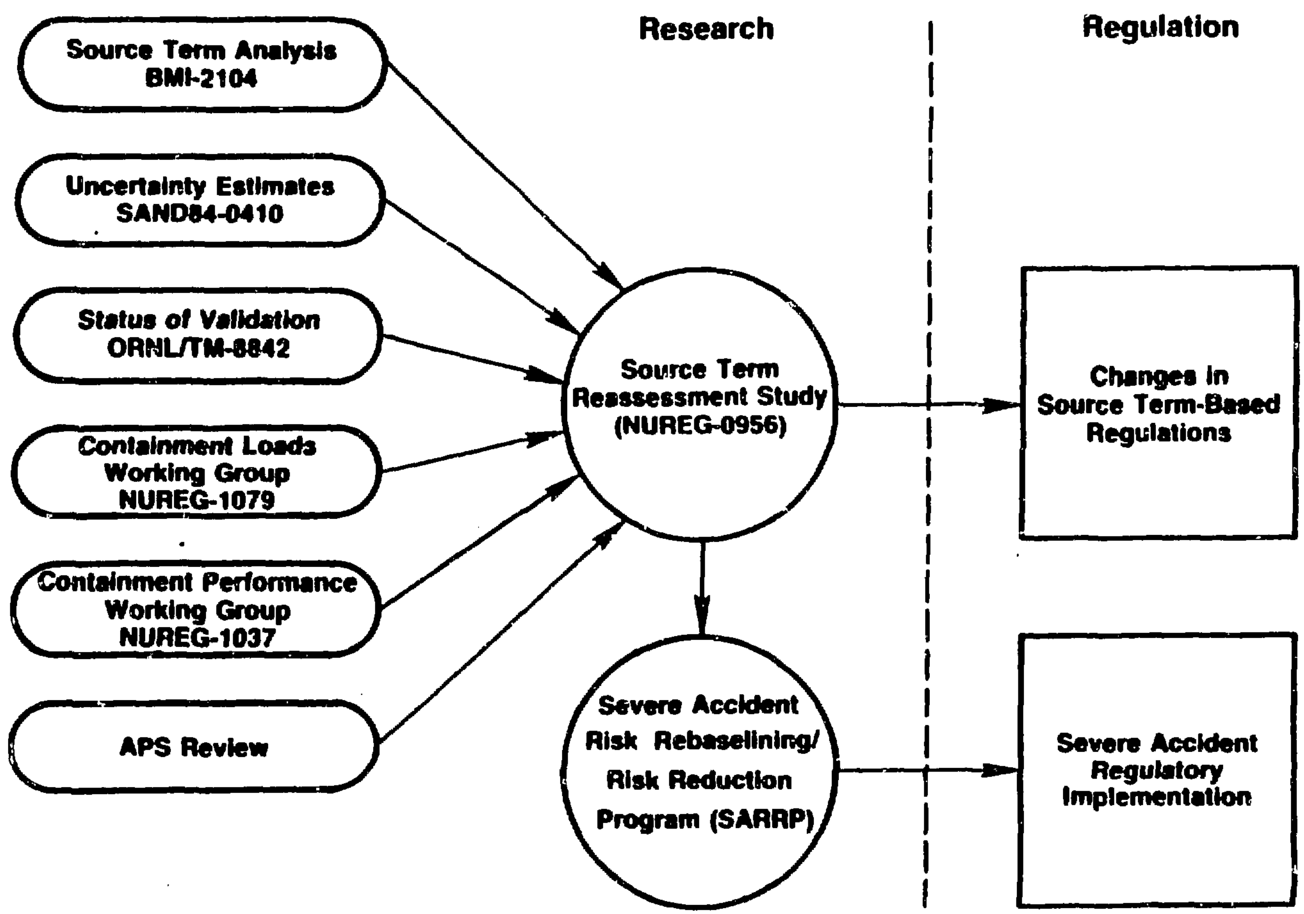
APS Conclusions

1. Considerable Progress Since WASH-1400. In a Number of Cases significantly Lower
   - Containments Are Stronger
   - Inclusion of Previously Neglected Physical and Chemical Phenomena
   - Inclusion of Additional Sites That Trap Radionuclides
2. Impossible to Generalize for All Sequences and Plants That Release Would Always Be Small

previously neglected physical and chemical phenomena. In fact, there are many we still don't include, because it makes it even more complicated. Current analyses also include many additional sites that trap radionuclides. That's a variation of the previously neglected phenomena.

Well, they made us a recommendation which of course we would like to follow although it makes it difficult. Don't generalize, don't do one PWR or one accident and say, "That represents the world." So when you hear someone tell you, "TMI showed us that iodine stays in containment; therefore you can change the emergency planning zone," don't believe him. I think you can change the emergency planning zone in the U.S., but it's not for that reason. You cannot generalize. It's too complex. You have to do a careful, systematic analysis of all of the accidents, sequences that count, all the significant ones, and look at the balance of results. You can't take one radionuclide and one accident sequence like TMI and go make regulatory changes based on it. What we are doing now in our regulatory process, which I think would be of great use to other nations, is shown on this program-relationship chart. Our source-term analyses are in this report, Battell Memorial Institute 2104. The uncertainty estimates, or I prefer sensitivity analyses, are in a Sandia report, SAND-84-0410; that is sometimes called the QUEST study (Quantitative Uncertainty Estimation for Source Terms). We have a very important one here, the Oak Ridge National Laboratory report, the "Status of Validation." What we're doing there is asking, very carefully, for each code, "What is the scientific basis for this code?" "How do I validate that this code predicts physical behavior?" That's an Oak Ridge Report. For containment we have two Nuclear Regulatory documents. We divide it into what we call "containment loads", that's the pressure and the temperature challenge to the containment, and "containment performance", this is the structural engineer saying, "Well, if you get that hot or that high a pressure, it'll break over here or leak over there." So we have these Nuclear Regulatory documents that describe our state of knowledge and how we would predict containment behavior. Lastly we have the APS review, which I just covered. All of that comes together with the NUREG-0956 we've just described.

Program Relationships



We are using this methodology now to do these integrated calculations of risk of different reactors. And we go in two directions then: Regulations, what do we do? First of all, we can look at things like emergency planning zones and use the new information to reevaluate and say, "Do we still think this is a sensible distance or approach?" We can also go into our regulations where we calculate the need for a charcoal filter. We have activated charcoal filters all over our power plants. Most of the science says there is not any iodine there for those filters to capture. The iodine comes out as a particulate, not as elemental iodine, and therefore it's not the charcoal filter that catches, nor the thyroid, its the HEPA filter. It's the High-Efficiency Particulate Filter that's going to catch it. So our whole design approach in the U.S. has been skewed, has been biased toward iodine control, and we have to reevaluate that.

Our results and the comments we have had already and on NUREG-0956 show that we need to be very careful about plant-specific review. The industry knows that and we know that. We're working with the U.S. industry for the most effective ways to do plant-specific risk analysis. It would be far too costly for the U.S. to turn around and say, "I want a level-3 PRA on every U.S. plant." Even though we have about 25 PRAs, that would be about 100 more. That's an awful lot of work and would take an awfully long time. No, what we are looking for are more effective ways to separate the differences from the similarities, so that, if we have a PRA of one type plant, we may learn enough to apply it, with corrections, to a similar plant. We're deep in that work right now. That's the work I referred to here, with the industry degraded core group.

There is another U.S. review that might be of interest to you. We have issued a "Severe Accident Policy" statement. The NRC issued that recently, also, just about a month ago. The statement does not come out so clearly, because it's filled with adjusted language. But the basic statement goes like this: We have looked carefully at the severe accident risk of current light-water reactors and, if you are sure that they don't have outliers (outliers are unique vulnerabilities, unique problems that we have found in many, many cases where in that plant, the peculiar way it could flood or a system is connected could be a high risk. And they're not general things, they're specific), as long as you can find the specific problems, these plants are safe enough. Not only are these plants, these light-water reactors safe enough, we are willing to license more of them. The way we would license them is a review of standard plants. We don't want to chase individual designs any more. Legally, we cannot forbid individual design. In this country, our laws require us to review any individual reactor design submitted. What we're trying to do is go as far as the law allows to tell George, over here, if he comes in with an individual design, he will get a very slow review. If he comes in with a standard plant, he moves to the head of the class. We're stretching our law there, but we want standard plants, we want to review them against our current, most up-to-date standard review plan. We want to consider the unresolved and generic safety issues in them, and we want to systematically consider their risk. With that kind of standard plant and that kind of careful review, we can say, "I have no reason to doubt the validity of licensing a plant like this." The first one we've done is GESSAR-II. That's the General Electric BWR-6 that I showed you at the very beginning, and Perry is very close to it -- the Perry nuclear plant. It's a refinement of that design.

When we go into our work, for some years now, we use risk analysis to decide the priority of work. How urgent is something? So we're now using new information, slowly but surely, in our best estimates of risk, for deciding if we should change something in an existing plant, a so-called back-fit. This is what we use to decide, "Is it worth fixing? How much will it cost to do it, and what improvement of public safety is achieved by doing it?" There's no sense using a silly estimate of risk. You want to use a good estimate of risk. You want realism, because you know the costs are real. We have priorities for addressing safety issues. We want to describe accurately the risk to the public. We have emergency planning and siting. I think it's worth saying here, our siting regulations go back to the mid-1970s. About 1979, we had a siting

RESULTS OF SOURCE TERM WORK

- o CURRENT PUBLICATION PUT SCIENTIFIC WORK IN PUBLIC VIEW
- o RESULTS SHOW NEED FOR CAREFUL PLANT SPECIFIC REVIEW
- o U.S. REVIEWS OF NEW AND EXISTING PLANTS
  - GESSAR II
  - EXISTING PLANTS WITH IDCOR
- o REVISED ASSESSMENTS OF RISK

report that said we really ought to revise our siting regulations. Many people, especially in foreign countries, thought that the U.S., which already had conservative siting requirements, was going to make them even more conservative. That's not true. What we did in 1980 and '81 was all of the technical work. We even proposed a regulation to the Commission, a regulation based on the old source terms -- WASH-1400. And the Commission said, "This is silly. We're doing all this work on new source terms. Suspend the activity on siting and wait until we have the new source terms and then come back to us."

You should appreciate that you can go look in our records and find the 1981 preparation of source terms, and the regulation was the same as the old one. It says the old one is conservative enough. In fact, if anything, you could justify even higher population density. So what we said is 500 people per square mile is an appropriate population density to use as a screening criterion. If you have 500 people per square mile, forget it. Don't worry about it any more. If you have <u>more</u> than 500 people per square mile, do an environmental appraisal. It's not prohibitive; you should compare the sites and determine which one in all of its factors is better. With the new source terms, that would be emphasized even more. It's not a high priority in the U.S. to have a new siting regulation now. When it does come, it will either be essentially the same as the old siting regulations, or it will be more relaxed. However, it's hard to make it easier in the U.S. because we have so much land. There's little incentive to make it easier.

Now, we have other things that are a real problem to us, because every day, in some power plant or other, I am notified of a problem: that they discovered that something isn't qualified to the TID source term, or the filter is giving problems. Charcoal filters are very difficult; the charcoal is difficult to test; it tends to channel in the beds. And here I am, about to shut down nuclear power plants in order to chase what I call a phantom radionuclide, elemental iodide. We need to change our regulations; it's quite complicated. We need a whole new thought process there. Containment leak testing: if you look in our regulations, we have a very peculiar way of implementing the TID siting

OTHER REGULATORY USES

* Best Estimates For Risk
    - Backfitting
    - Issue Priority
    - Environmental Statements

* Site Related Issues
    - Emergency Planning
    - Siting

* Equipment Issues
    - EQ Source Term
    - Choice Of Instrumentation To Follow Accidents
    - Filters
    - Containment Leak Testing

calculation, the one I showed you on the slide earlier. In that calculation, you assume a containment leakage. That containment leakage is put into the technical specifications for the reactor, and then we have a regulation, 10-CFR-50, Appendix J, that tells you how to test to prove that you have no greater than that containment leakage. It's a very costly, frequent test of the reactor and its penetrations, and it really contributes very little to public health and safety. We know now that what we need to do is change that whole system of regulation. It's very much like the charcoal filters. It is costly; it is difficult; and it really isn't worth it. So a very different way of containment integrity or leak-tightness testing will be developed. However, those are difficult and they will take time to do.

Now let me turn to safety goals, because, as I said earlier, we are always asking ourselves, "Is this safe enough, or should I do something?" We have been evaluating quantitative safety goals, using PRA numbers, for several years. For your convenience, I listed here the NRC report number, NUREG-0880. It was published in 1981 or 1982. The philosophy of the safety goal is this: I will look at any individual member of the public who lives close to a nuclear power plant. And I want to be able to say that the risk of accidental death or of cancer death to which that person is exposed due to the nuclear power plant is so small that they shouldn't have to think of it. That's the ideal way to regulate. We regulate aircraft safety. This is a bad year for aircraft safety, as you know from reading the news, and you all have to fly home. But it is still a safe way to travel and the objective is to regulate aircraft safety so that a member of the public can buy the ticket on the basis of convenience and cost, how nice a dinner you get on the airplane, the seat, the movie, the things that you can easily judge for yourself. You should not have to judge the relative safety between a DC-10 and a 747. That's not appropriate. So safety should be regulated so that the risk is negligible against that background. So what we chose was: If accidental deaths and cancer deaths are increased by no more than 1/10th of 1% to the people close to the power plant, that is negligible. And that's rather interesting, because in your everyday life,

SAFETY GOALS

- NRC EVALUATING QUANTITATIVE GOALS (NUREG-0880)
- BASED ON NEGLIGIBLE INCREASE IN RISK FOR THOSE INDIVIDUALS CLOSEST TO PLANTS
- FOR ACCIDENTAL DEATH:
  $0.001\ (5 \times 10^{-4}/YR) = 5 \times 10^{-7}/YR$
- FOR CANCER DEATH:
  $0.001(1 \times 10^{-3}/YR) = 1 \times 10^{-6}/YR$
- FOR CORE MELT: $1 \times 10^{-4}/YR$

whether you know it or not, you change your accidental death risk easily a factor of 2 or 3 from one thing to another, just by changing where you live, changing where you work, changing many different things changes risk. In fact, in the U.S. variations are far greater than a factor of 10. Therefore, it is legitimate to say 1/10th of 1% is truly negligible. I often say, "100% is arguable and possibly defensible, as a change. 10% is certainly defensible. 1% is trivial, and 1/10th of 1% is truly trivial." But that's a good place to be. If you can get there economically, that is a good place to be, as a regulator, because you have margin of safety. I would point out to you that on this slide I have a typographical error. For cancer death in the parentheses is 10^{-3}. It should be twice that. Of course, the product should then be 2 10^{-6}. Those are the figures that are a national average for the United States. They will differ a little bit for other countries, but that's not significant. At 1/10th of 1%, feel free, use our safety goal. The margin avoids any question about change in those numbers. We are using those numbers, derived that way, as a test. We're wondering whether we should formally use them. Then, for core melt, I defined it as the conditions leading to severe core damage, we use 10^{-4} per year.

Now, I would just give you, on the next slide, a feel, for where our best state of knowledge says we are relative to these safety goals. If you look at the recent PRAs, based on WASH-1400 source terms, we find that core melt is equal to, or four times or so greater, than the goal of 10^{-4} per year. The prompt death, the high dose-rate death, is at 10% to 40% of the safety goal, so it's close. The latent death is generally much lower. That's the risk we get when we use current PRAs of existing plants in the United States with WASH-1400 source terms. Now, if we look at a new plant with new source terms, and we had that first example I cited, GESSAR-II, the results are much lower. So our analysis is that the core melt is at least 10 times lower than the safety goal. The prompt death is essentially zero. You don't calculate any. The latent death I give it as an approximation of 0.001 of the safety goal. It's probably even lower than that. The whole point, and I said this to our Advisory Committee on Reactor Safeguards earlier this month on the

COMPARISONS TO GOALS

BASIS	MULTIPLE OF GOAL		
	CORE MELT	PROMPT DEATH	LATENT DEATH
RECENT PRAS WITH WASH-1400 SOURCE TERMS	1-4	0.1-0.4	0.001-0.05
NEW PLANT WITH NEW SOURCE TERMS (E.G. GESSAR II)	0.1	≈ 0	≈ 0.001

- o PUBLIC SAFETY IS PRESERVED
- o ECONOMIC RISK IS IMPORTANT (CORE DAMAGE OR MELT)

GESSAR meeting, the public safety is clearly preserved. We are not even close to the range of concern in a safety goal. We are well below that range, and that range is clearly acceptable. So the public safety is clearly preserved, and I want to keep it there.

We can now turn our attention to the economic risk. And remember what I said at the beginning. In English, it comes out a little bit better. The owner's investment is hostage to the public health and safety. There's no way to hurt the public without first bankrupting the owner. You have to hurt the core. You have to cost the owner of the plant at least a billion dollars in order to begin the threat to the public off-site. So that's why we now have a regulatory system where, with much better understanding of reactor safety, of reactor risk, we can do cost-benefit analysis and say when is safety change worthwhile? At the same time, the question of economic risk is there and we have a legal problem: Do I regulate the owner's reliability? Do I regulate the off-site damage after the people are gone? That's a difficult thing, and each of you in your own country will answer that in your own terms.

IX OVERVIEW OF THE PROCESSES OF RELIABILITY AND RISK MANAGEMENT

Edwin L. Zebroski
Chief Nuclear Scientist
Electric Power Research Institute

OVERVIEW OF THE PROCESSES OF MANAGEMENT OF PLANT RELIABILITY AND RISK

Dr. Edwin L. Zebroski
Chief Nuclear Scientist
Electric Power Research Institute

Nuclear energy is unique in three fundamental ways from other industrial activities, and particularly from other power production activities. The most obvious one is that all of the nuclear activities in the world are - in a sense - linked together. If you have either a very bad experience or a very good experience it affects everyone else in the world. So if you are to do a good job you have to have an interest, you have to maintain an interest in other peoples' experiences and how they influence you. At one end of the scale, a few plants worldwide, and perhaps the top ten in the United States are turning in amazingly good performances, capacity factors well over 80%, availabilities in the range of 90%, well beyond the original design expectations when these plants were put together. At the other end of the scale there is occasionally a very troublesome event, either an accident or what is perceived to be a near miss to an accident, which then frightens everybody or makes concerns for everybody. So we need to be aware of both ends of this scale.

In looking at the speakers that we have lined up here, they represent something like 400 man-years of experience with nuclear power plants either in design, operation or construction. At least a few of the speakers who have hands on experience in either running or designing plants represent the most successful end of the spectrum of experience. I'll talk more about that distribution problem later. Today I'm going to go through a synthesis of what risk management or safety management, if you want to be more optimistic about it, is all about. It is much more than any one of the disciplines that we think of in this connection. We start out with a very encouraging first chart here. The world experience is now over 3000 reactor years of operation of commercial power units. There's also over 3000 years of military reactors - for which we don't have shared experience - but, the occurrence of really serious accidents in the sense of threatening the public is also very very low. It's distressing to hear today that 500 people are killed in an airplane and that only gets on the fifth page of the New York Times or other newspapers. If one person is killed in a nuclear power plant it will surely hit the front page of every newspaper in the world. So there's a great difference in perception. Nevertheless, the real record is very good.

Actual radiation releases, as far as we know, have been miniscule so far. The criteria that I have for de minimis or negligible is that the radiation given to any member of the public is of the same order of magnitude as the natural variation in background that a person is exposed to in any given year. As you may know, I was one of the people that was asked to organize a technical support group at Three Mile Island immediately after the accident there. One of the things we did very thoroughly was to analyze the radiation exposures resulting from the gas release that occurred. When all was said and done, the "fence-post dose" at the site boundary was of the order of 20 millirem with a possible localized point location somewhat higher. The highest population exposure (a hypothetical somebody who had spent 24 hours a day in the open for two weeks at the fence) might have been of that order. The actual doses are undoubtely much lower. The interesting comparison is with the dose rates from nature, (I have a map of the dose rates from nature in the vicinty of Three Mile Island) are a function of position. The lowest doses happen to be along the river, around 50 millirem per year. If someone evacuated and went to Uncle Joe's farm, a few miles away, typically they would see 75 or 100 millirem per year and their exposure would actually have been higher than if they had stayed next to the plant! We know that, but the public doesn't know or believe that. The public tends to believe much more unpleasant things. There's a great mythology that plants and animals died and children were born defective and so on, but certainly there's no scientific basis for that. So what we're talking about on radiation is largely occupational exposure, and that is a matter of management.

I'm sure you all know what a feedwater transient is. The next slide is a distribution function for the frequency of feedwater transients for a particularly period of time, the number of transients per thousand hours.

WORLD NUCLEAR PLANT SAFETY RECORD

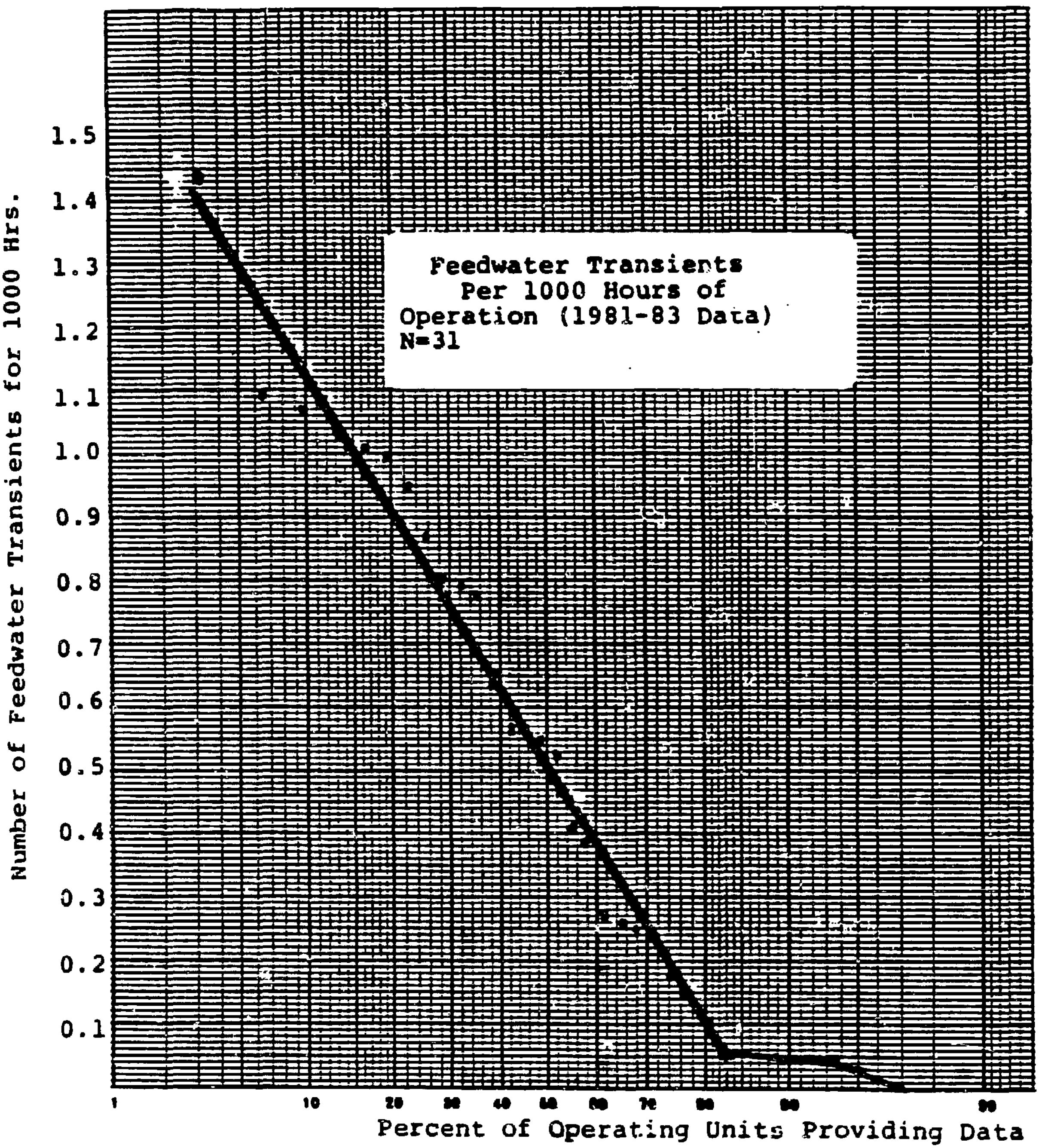
- THE EXPERIENCE TO DATE COVERS ABOUT 3,100 PLANT-YEARS OF OPERATION FOR COMMERCIAL (NON-MILITARY) POWER UNITS

- TWO KNOWN INSTANCES OF SIGNIFICANT RADIATION RELEASES TO ENVIRONMENT IN ACCIDENTS
    - WINDSCALE (1958)
    - THREE-MILE ISLAND-2 (1979)

- THE HIGHEST EXPOSURE TO NEARBY PUBLIC FROM THESE ACCIDENTS WAS EQUIVALENT TO LESS THAN ONE YEAR OF NATURAL RADIATION BACKGROUND, AND POSSIBLY LESS THAN ONE MONTH OF BACKGROUND

- NO KNOWN BIOLOGICAL EFFECTS HAVE BEEN OBSERVED FROM SUCH EXPOSURES; THEORETICAL CALCULATIONS SUGGEST THAT THERE COULD BE AN INCREASE IN CANCER INCIDENCE (FOR TMI-2 ACCIDENT, ZERO TO TWO CASES ADDED TO ~ 300,000 EXPECTED NORMAL LIFETIME INCIDENCE)

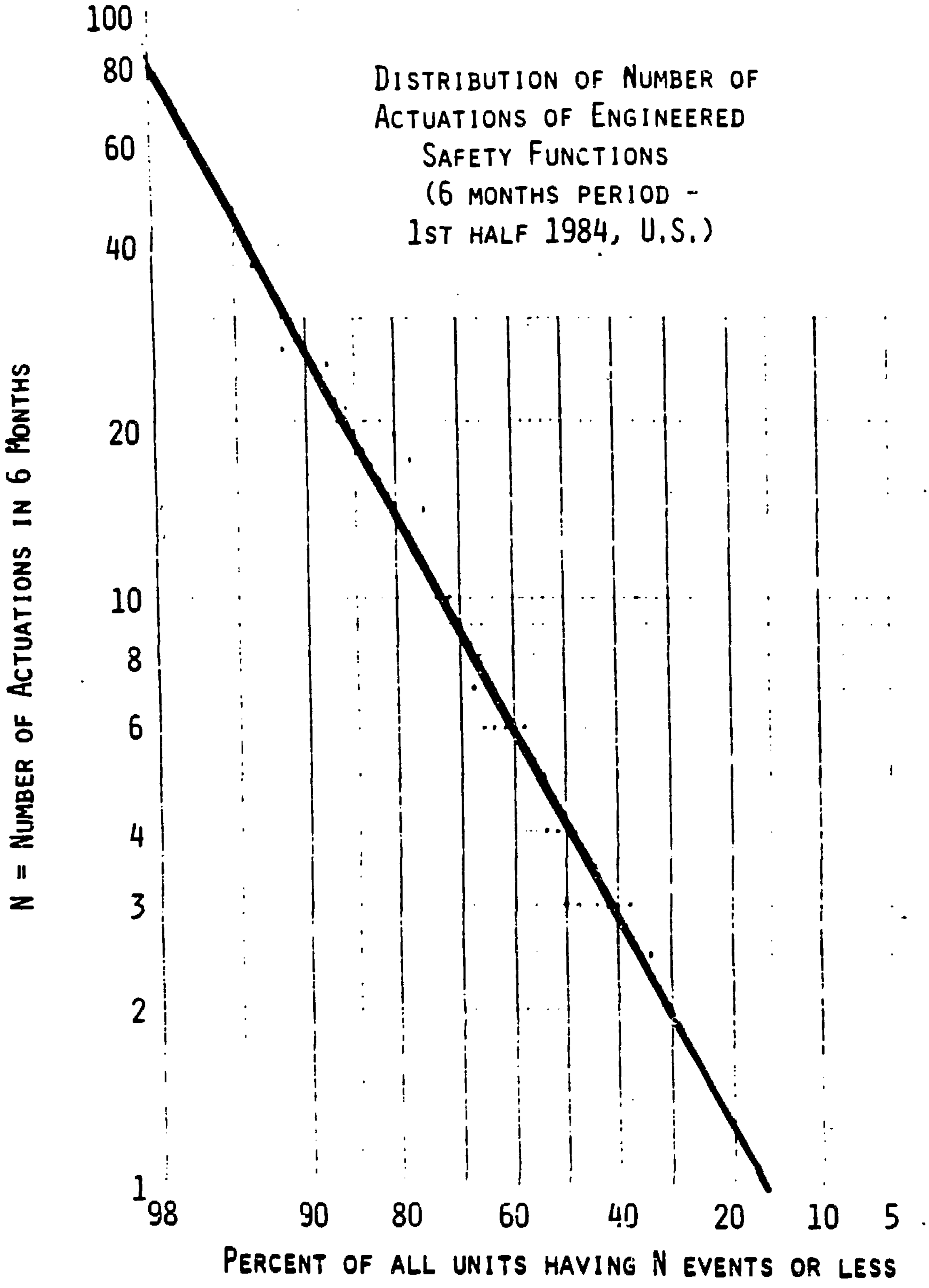
- THE OCCUPATIONAL EXPOSURES INVOLVED IN PLANT OPERATION AND MAINTENANCE ARE MUCH LARGER THAN PUBLIC EXPOSURES (EVEN INCLUDING ACCIDENTS). EXPOSURE MANAGEMENT TECHNIQUES ARE AVAILABLE THAT MINIMIZE SUCH EXPOSURES


Feedwater Transients
Per 1000 Hours of
Operation (1981-83 Data)
N=31
Number of Feedwater Transients for 1000 Hrs.
1.5
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
1
10
20
30
40
50
60
70
80
90
Percent of Operating Units Providing Data

And this is from the population of operating reactors in the United States. This is a probability distribution plot. The median is that there is a little less than 0.5, feedwater transients in 1000 hours of full power operation. That is typical. If you look at the range of the distribution, 20% of the plants that somehow manage to get down in the range of one transient in 10,000 hours or even 20,000 hours. There is a little magic, or mystery or know-how of management that is showing here. On the other end of the scale we see more than one transient per 1000 hours, or over ten times the rate of the best 20%. There's something to be learned from both ends of the scale. But the distribution is important. If you look only at the averages, you lose the important insights from the distribution function.

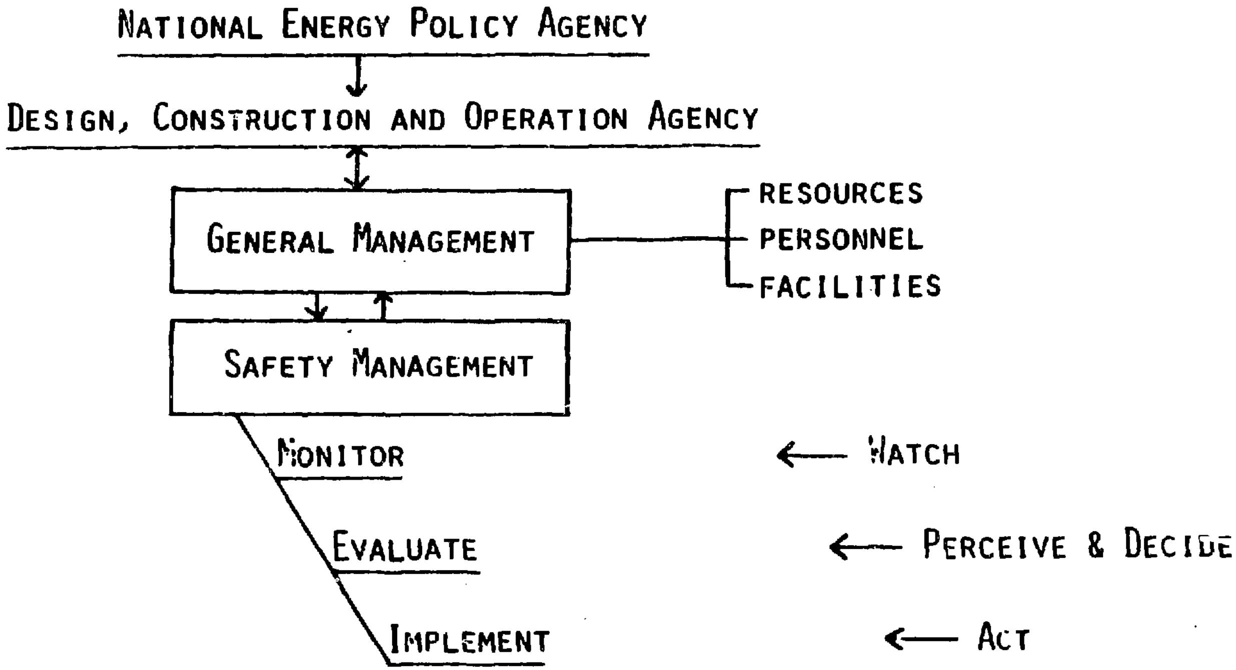
This next slide is more complicated; this is a lognormal plot, showing the number of engineered safety function actuations in a six month period. This is a total of all engineered safety functions. The range here is really astounding. Here you have again about 20% of the population has only about one such challenge to safety functions in a 6-month period, one or less. Again, about 20% of the managements and operating people somehow have the magic to avoid tripping the turbine and tripping the control rods and tripping the feedwater system. At the other end we see ten percent of unfortunate plants that experience 25 to 80 such events in this period of time! So, if anything else doesn't convince you of the importance of safety management, I hope these charts do. In other words, it's a matter of choice where a plant or a country operates on this scale.

Now we shall talk about the general conditions for a good safety management. The main theme of this part of the talk is that it requires an orchestration of many disciplines. Perhaps that is self-evident. More than just the orchestration, it requires the orchestration with the proper weighting of the different activities. There is a national interest in making energy. This leads to an agreement to produce it by nuclear means, and the means are allocated to build the plants. Then you get down to the hardware of actually specifying, designing, building, and running. It is always the main function of general management to marshall the facilities and people. We should put these two blocks together. Safety management and



DISTRIBUTION OF NUMBER OF ACTUATIONS OF ENGINEERED SAFETY FUNCTIONS (6 MONTHS PERIOD - 1ST HALF 1984, U.S.)
N = NUMBER OF ACTUATIONS IN 6 MONTHS
100
80
60
40
20
10
8
6
4
3
2
1
98
90
80
60
40
20
10
5
PERCENT OF ALL UNITS HAVING N EVENTS OR LESS

SAFETY MANAGEMENT FUNCTIONS



general management cannot be separated. There's a very simple algorithm. I was for some years with the Institute of Nuclear Power Operations and one of the jobs we did was to visit plants with a team of about 15 to 20 people. Perhaps half of them would be actual plant superintendents or reactor operators and the other half would be analysts or engineers. They would examine the plant and its records for two weeks, and then make recommendations to the management of things that they might want to improve. They are also told of the things that they were doing very well - good practices that might be imitated by others. The interesting thing is that the plants that generally got a high score were those where safety management and general management were hand in glove; there was no separation; there was no communication barrier. There was no question of just a meeting and a report once a month; it was a daily ongoing interaction. If this function is separated very much from the actual operating or construction management then you get difficulties simply from communication problems. So the rest of my talk will basically be about how and where these three functions are done. Tou will hear more in some of the other talks on the function of plant evaluation. Professor Pigford especially will talk about diagnostics; some of the other talks will be about probabilistic risk analysis.

The management function is to watch, to monitor, to evaluate; to understand what you're seeing, and to make decisions about it, and then to implement changes where they are needed.

One of the very difficult things in a new program, but also in an old program of nuclear energy, is to say where is the responsibility centered? You have this tremendous overlap. It's very easy to for people to say, "He's responsible for that, I'm not responsible." So a very important thing is to understand that there is this great set of relationships involving the government, the safety regulatory authority, and usually some kind of a board of directors or board of control. These bridge from the national level to the operating level. The administrative apparatus finally down to the local level of the plant superintendent and to the department and shift supervisors. If most of the plant safety analysis and awareness occurs only at the local level you don't have good communication up to the level of people who can commit resources to sign the orders and bring in the people and the equipment, then it becomes very difficult to get good implementation of safety management. The definition of those roles is a particularly important function.

Some of the responsibilities can be self-defined and they are culture-independent. Someone may say "we do things differently in our country", and to some extent that is always true. The functions will have different titles, they may have different locations, but the functions must be provided in one way or another, with different titles. You can discover where some of these responsibilities are in two ways: either there's no other place it can happen, or - it can't happen there. For example if a detailed design improvement of a feedwater system is under discussion, it cannot happen in the capital of the country or in the headquarters of the regulatory authority. But the regulatory authority can decide that that is a necessary activity and make sure that somebody is doing it. You can define, at least in the broad terms, where these responsibilities go by simple inclusion and exclusion reasoning. The key point is to keep that map reasonably coherent so there are not gaps between responsibilities. (Gaps means a situation in which people think that someone else is covering the problem, but no one is.)

The function of board of directors or board of control, if it's at a national authority, is especially important. I've attempted several times to enunciate what I call the first and second law of responsible risk management. The first law is to make sure that there are enough resources to do the job right. The second law, which is even more important, is that the priorities allocation of resources is correct. If you misallocate resources you clearly reduce safety. This theorem can be proven both by experience and in several theoretical ways. If you put too much effort on a small problem, you starve a bigger problem of effort and overall safety suffers. The essential role of national allocation of resources means that the communication to the people who finally appropriate or collect the money and allocate the resources is a crucial aspect of good safety management.

SAFETY MANAGEMENT

<u>WHO IS RESPONSIBLE FOR WHAT?</u>

- <u>SAFETY AUTHORITY</u>
- <u>BOARD OF CONTROL</u>
- <u>CHIEF EXECUTIVE OFFICER</u>
- <u>CHIEF NUCLEAR OFFICER</u>
- <u>CHIEF ENGINEER'S OFFICE</u>
  - SAFETY & RELIABILITY ANALYSIS GROUPS
- <u>PLANT SUPERINTENDENT</u>
  - CONSTRUCTION
  - OPERATION
  - PLANT SAFETY
  - HEALTH PHYSICS
- <u>DEPARTMENT AND SHIFT SUPERVISORS</u>
  - OPERATIONS
  - MAINTENANCE
  - TECHNICAL SUPPORT

OVERSIGHT FUNCTIONS HELP TO SET THE CLIMATE FOR SAFETY AND PRODUCTIVITY

REGULATORY AUTHORITIES

- LICENSING
- STANDARDS AND GUIDES
- MONITOR COMPLIANCE
- ALLOCATION OF RESOURCES

BOARD OF CONTROL OR DIRECTORS

- RESOURCES
- TOP PERSONNEL
- MONITOR SAFETY AND PRODUCTIVITY PERFORMANCE
- EFFECTIVE ALLOCATION OF RESOURCES

A regulatory authority can only set the general climate for doing things right and provide the oversight that they are happening properly. One of the most difficult functions, especially from a regulatory standpoint, is that an effective operating management must integrate a great many different activities. There are some easy symptoms to tell whether these functions are working together well. The most obvious one is communication. If you have a situation where one department doesn't talk to another, or headquarters doesn't talk to the plant, you know that some things will not be done very well. Poor communication is the most obvious symptom. So one of the key functions of the higher levels of management, both on the regulatory side and on the operating side, is to ensure that there is open and free communication. If two people are not very cooperative and they don't talk to each other, you're going to have to change one or both of them, - either psychologically or by job position.

Coordination of goals in schedules. If the construction people have one goal and the operating people have a different goal and the regulatory people have still a third goal, things are not going to go very well. The coordination of goals again is a matter of communication and negotiation. The top management, in the regulatory side, the government side, and the operating organization must orchestrate these relationships. Even if the operating organization is a government organization, it doesn't matter. These interfaces are still there and must be nurtured. Finally, the operating management, on one hand, and the regulatory authority on the other hand, must also integrate with the government in general and with the public. The public doesn't vote on these issues in any direct way, but in a very indirect way it does set the climate for what can and should be done.

The "How" of the integration of safety management. The top management has the control of the resources. The two basic things that are to be integrated are; (1) the actual safety analysis, which provides the intelligence, the sense of direction, and what it is you're going to do, and (2) the resources to get them done. The analyst can't do that alone and the management can't do that alone. Here is a list of the kind of resources that a plant management has available. Perhaps the most basic, other than

INTEGRATING ROLE OF MANAGEMENT

MANY FUNCTIONS MUST WORK TOGETHER SMOOTHLY TO ACHIEVE SAFETY WITH PRODUCTIVITY

- COMMUNICATE PROMPTLY
- COOPERATE FREELY
- COORDINATE GOALS AND SCHEDULES

TOP MANAGEMENT MUST ORCHESTRATE:

- COOPERATIVE STYLE OF ORGANIZATION
- OPEN COMMUNICATION
- TIMELY AND INFORMED DECISION-MAKING
- CHANGES IN STRUCTURE OR PERSONNEL WHEN NEEDED
- ADEQUATE RESOURCES
- GOOD RELATIONS WITH GOVERNMENT AND PUBLIC

(THESE ROLES CANNOT BE DELEGATED!)

FUNCTIONS OF SAFETY MANAGEMENT

<u>INTEGRATION OF INFORMATION AND RESOURCES</u>

<u>SAFETY ANALYSIS AND CONTROL</u>

<u>MANAGEMENT RESOURCES</u>

- ORGANIZATION
- PLANNING AND SCHEDULING
- CONSTRUCTION AND OPERATION FOLLOWING
- INFORMATION HANDLING

FUNCTIONS OF SAFETY MANAGEMENT

MANAGEMENT RESOURCES

ORGANIZATION

- STAFFING SELECTION
- TRAINING AND REQUALIFICATION
- PERFORMANCE MONITORING

PLANNING AND SCHEDULING

- OPERATION
- MAINTENANCE
- TESTING
- EMERGENCY DRILLS

CONSTRUCTION AND OPERATIONS FOLLOWING

- TRACKING PRACTICES & MONITORING
- QUALITY ASSURANCE SYSTEMS
- BACKFIT AND REWORK CONTROL
- CORRECTIVE ACTION TRIGGERS

INFORMATION HANDLING

- MANAGEMENT INFORMATION SYSTEMS AND TRACKING
- PLANT DESIGN INFORMATION
- CODES, STANDARDS AND REGULATIONS
- VENDOR INFORMATION
- INFORMATION SYSTEMS OPERATION
- COMMUNICATIONS
- REAL-TIME MONITORING SYSTEMS

the command and control structure itself, is the handling of information, the tracking of needs, decisions, and implementation. A power plant involves today an enormous amout of information. Typically, of the order of a million records are required in U.S. system to be maintained at archival level. (Ten to 20 million over plant lifetime.) That means you must be able to go back and find how something was built or designed or how it was calculated. The information handling is very important. It should be self-evident that a capability for timely - that is real-time monitoring - is essential. The idea that you get a report once a month or once a quarter and then make decisions is grossly unrealistic. If you want to maintain safety, reliability, and productivity, real-time control of information is needed.

The basic issue of the safety analysis. There are three basic functions: monitor, evaluate, and implement. The chart lists some of the resources that you require for safety analysis. The most basic thing is the input of information. You have to able to monitor, not only how your own plant works, but how similar plants of the same design type work. I have been involved in failure analysis for a number of years. I find one of the surprising things to be that if a particular component or system has a failure mode in one design it will very often show a similar failure mode in another design. The idea that another plant was designed by Combustion or Westinghouse and this one is designed by KWU or GE and therefore "does not have that problem" is often not valid. You usually should say, "If there's a similar component or system in this plant, I often can learn something even from a plant of a different nameplate." There are now information systems that make it possible to do this very easily.

The safety evaluation process. The next chart is a schematic of the safety evaluation process. First of all you have to perceive a need to do something, or potentially a need to do something. You had an event or you had a quality inspection or you've had an observation or an experience in another plant, which causes you to ask, "How does that affect me?" So that raises a question of an importance judgment. Even before you do any analysis you have to make a preliminary judgment of the importance of that issue. That can really only be done with a great deal of experience. Many

SAFETY EVALUATION PROCESS

EVENT OR DEFICIENCY OBSERVED
OR
POTENTIAL IMPROVEMENT PERCEIVED
OR
EXPERIENCE OR ANALYSIS ELSEWHERE

↓

IMPORTANCE JUDGEMENTS

↓

- <u>DETAILED ANALYSIS</u>

  - THREAT LEVEL?
  - OPTIONS FOR PREVENTION?
  - OPTIONS FOR MITIGATION?
  - INTERIM MEASURE NEEDED?
    - ALERTING PEOPLE
    - PROCEDURE CHANGES
    - TRAINING CHANGES

↓

- <u>DECISIONS FOR CHANGES NEEDED</u>: <u>YES OR NO</u>

  <u>IF YES</u>: WHICH OPTION?
  WHAT TIMING?
  WHAT RESOURCES?
  PLAN AND SCHEDULE?

↓

- <u>IMPLEMENTATION MONITORING PLAN</u>

of you know that there is something called the "Significant Events Program," which is carried on now in Atlanta. There are about 50 people that perform that function. Typically, when an event occurs, it is reviewed by two people who have operated or supervised the same plant or a plant of very similar design, and one person who is a design analyst. The three of them, just by engineering judgment, with no mathematical analysis, get together and ask if this thing is significant or if it is small enough that it can be cured locally. That preliminary judgment of importance is a very critical step. If it is judged that it might be important, then it is subjected to detailed analysis. The threat analysis means: given that this event has happened, or this deficiency exists or this component is not functioning very well. What are the possible damaging consequences? Say I have three diesels and one isn't running, should I run the plant or not? should I fix it now - or can I take a week, or a month. The threat level is where the probabilistic analysis, helps to make those kinds of discriminations.

Given that you perceive that there is something worth correcting, you ask, "Well what are my options; how can I fix it?" Typically, you have three different ways to go. You can prevent, that is increase reliability or redundancy. In the NASA reliability program, that's their primary tool. If they see that a particular component isn't quite reliable enough, they put in two of them. If that's not good enough, put in three. The working of that process was very visible in the shuttle flight prior to the Challanger - where one of the engines malfunctioned while they were still boosting to orbit. Even with one engine partly out, they had enough redundancy, enough duplication margin, that they were able to finish the mission.

Mitigation. However, there are some things for which redundancy is not enough. Then the analysis tells you that the reliability must be extremely high, when a single failure can be disasterous. This defines the level of discipline of testing and failure analysis required to perform safely.

In the case of the Challanger disaster, the weakness of the joint design apparently has recognized to the extent that a revised design with was flawed, or poorly communicated, so that decisions for timely implementation did not happen.* The prevention of failure by increasing reliability, or increasing redundancy, is a very common way to go. However, no matter how well you prevent, you must also think, well what if it happens anyhow? And then you talk about mitigation. How can I cope with this failure or malfunction, even if it occurs? And then finally, most serious mitigation or prevention activities take quite a while to implement. If it's a design change, it may take one or two years, or to the next shutdown. If you take the luxury of shutting the plant down every time you see a deficiency, you will be down more than you are up. So you have to make an importance judgment. At some point you may hear the phrase, "time integral of risk." In other words, you can let a small risk, run for a long time, whereas a large risk you'll fix quickly. That's a more subtle point, which I'm sure will be talked about later. But you always have an interim way of improving. That is to alert all of the people involved that this is a potential problem. If it requires operator action or maintenance action or increased surveillance, people can be alerted that they should be more careful and thorough on that system. As you know, after Three Mile Island the reliability of feedwater systems began to be studied very intensely. One plant, the Ooonnee Plants 1 and 2 were operating at that time. They were allowed to continue operating even though they had this design deficiency. Th difficulty was temporarily overcome by stationing an operator at each feedpump with telephone communication to the control room. If one of those feedpumps stopped, the corrective actions could be taken without waiting for the safety systems to decide that the system was going dry, going over-temperature, losing water levels and so on. The corrective action could be taken before any of that started to happen. That was considered an adequate and safe interim fix until the design control changes could be made.

You have to make decisions on <u>whether</u> you take a corrective action, and if so, <u>which</u> ones and <u>when</u>. As I've already indicated, the question of timing is very important. One very interesting observation is that a

* (Note added in proof September 1986)

number of utilities in this country and some of the utilities overseas have coped with backfits and improvements with very little loss of time in construction or operation. In other locations the same changes, not well anticipated, have caused great delays in either construction or operation. The question of timing is very important and that also goes back to good judgments of importance levels.

Resources for doing analysis. Let's consider the resources available when you're doing analysis. We'll talk about both probabilistic and deterministic analysis. Deterministic analysis is simply a highly generalized way of saying what you normally mean by engineering design or function analysis. Probabilistic analysis has a number of dimensions but the most fundamental one is to have the right picture of a potential occurrence, the right model of how A affects B, how B affects C and so on. One of the important uses of probabilistic analysis is to get an understanding of how a system behaves both in normal operation and in various transients. I'll talk about implementation criteria more later.

When we talk about engineering analysis we usually talk about deterministic analysis. The chart lists some of the classical categories of deterministic analysis such as thermal-hydraulic, structural, and neutronic. Obviously there are others: the control, shielding, people radiation, and so forth. On the same chart, we are interested not just in the steady state operation but in the transients and the upsets. Besides the steady-state structural analysis we need to know how does it respond under various kinds of operating transients, thermal shocks, or abrupt load changes and, seismic events. Not all of these are necessarily covered in the initial design analysis.

Data on experience. One of the most important perceptions that safety management should have, both at the plant level, at the headquarters level and at the regulatory level, is that any aspect of this business involves a pyramid of experience. At the tip of the pyramid is my opinion or your opinion of what counts on this particular problem. It's based on our own personal experience and operation and what we've read and what we've learned. We have the next level of experience, our immediate colleagues or operating organization, what their experience and knowledge covers. In the

INPUTS TO ANALYSIS AND DECISIONS

- **DATA**

  FAILURE STATISTICS
  SIMILAR EVENTS ELSEWHERE
  ANALYSES BY OTHERS
  LOCAL OBSERVATIONS

- **PROBABILISTIC ANALYSIS**

  RETURN FREQUENCIES
  CONSEQUENCES
  SEQUENCE VARIATIONS
  LOCAL VS. GENERAL STATISTICS

- **DETERMINISTIC ANALYSIS**

  SIMILAR EVENTS ELSEWHERE
  GENERIC ANALYSES
  DETAILED, CASE-SPECIFIC ANALYSIS
  SEQUENCE RISK ANALYSIS
  PLAUSIBLE VARIATIONS

- **IMPLEMENTATION CRITERIA**

  ACCOMPLISHMENT MEASURES
  COSTS TRACKING & CONTROL
  MEASURES OF EFFECTIVENESS OF REMEDY

Deterministic Analysis Inputs

Thermal-Hydraulic Analysis

- General & Local Flows
- Temperature distributions
- Hot Channel Factors
- Power Level Charges
- Thermal Cycling & Thermal Shocks

Structural Analysis

- Core, Fuel & Control Rods
- Vessel, Supports, Core Supports
- Primary Piping & Supports
- Pumps & Valves
- Steam Generator
- Containment Shell & Hatches
- Shielding
- Foundation & Base Mat
- Auxiliary Building
- Electrical Distribution
- Turbine-Generator-Condenser
- Cooling Towers & Water Intakes

Neutronic Analysis

- Power Distributions
- Control Margins
- Radiation Damage Doses
- Reload Patterns
- Stretch-out Cycles

DETERMINISTIC ANALYSIS INPUTS

THERMAL-HYDRAULIC ANALYSIS

- STEADY STATE
- NORMAL RATES OF CHANGE
- TRANSIENTS & TRIPS

- GENERAL & LOCAL FLOWS
- TEMPERATURE DISTRIBUTIONS
- HOT CHANNEL FACTORS
- POWER LEVEL CHARGES
- THERMAL CYCLING & THERMAL SHOCKS

STRUCTURAL ANALYSIS

- COLD STRESSED
- HOT-STRESSED
- THERMAL STRESSES
- TRANSIENTS
- SEISMIC

- CORE, FUEL & CONTROL RODS
- VESSEL, SUPPORTS, CORE SUPPORTS
- PRIMARY PIPING & SUPPORTS
- PUMPS & VALVES
- STEAM GENERATOR
- CONTAINMENT SHELL & HATCHES
- SHIELDING
- FOUNDATION & BASE MAT
- AUXILIARY BUILDING
- ELECTRICAL DISTRIBUTION
- TURBINE-GENERATOR-CONDENSER
- COOLING TOWERS & WATER INTAKES

MATERIALS

- SPECIFICATIONS
- PROPERTIES
- PROCESS CHANGES
- WELDS
- SUBSTITUTIONS

NEUTRONIC ANALYSIS

- STEADY STATE
- POWER CHANGE
- TRIPS
- TRANSIENTS - W/DELAYED SCRAM

- POWER DISTRIBUTIONS
- CONTROL MARGINS
- RADIATION DAMAGE DOSES
- RELOAD PATTERNS
- STRETCH-OUT CYCLES

nuclear energy business, at least initially, the conventional belief on balance of plant was that it's just like another steam boiler. It's just another heat source and the rest of the plant is conventional. That might be a tolerable perception if you have a low capital cost system that doesn't need a high operating factor and doesn't frighten people when it has transients. You want a high operating factor in a nuclear plant because you have a lot of capital tied up and because you have the very lowest incremental fuel cost of any power source that you have. Then the answer comes out differently. Even the balance of plant wants to have very sophisticated analysis, maintenance and occasional upgrading in operation. The importance here of this chart is that the data enable the analysis to be based not just on your own experience or the local experience or even the experience of that type design but actually on the whole world's available data base. That ability now is approaching. We now have really a quite integrated data base in the United States. There's the Significant Events Program data base, which basically has recorded and stored and analyzed and made remedy recommendations on all of the significant operating events since 1979. That is all on-line, computer searchable, available to every utility in the United States and in 13 other countries. That is not quite the world experience data base because it's mostly United States but I will observe that the United States data are still well over half of the total world's searchable data to date. The next biggest program in the world is less than one third as big as the United States and has fewer years of experience. However, fortunately we're also getting from 13 and perhaps 14 other countries the beginnings of inputs of this same kind of significant information in a conveniently accessible data base. IAEA has added to that now with a parallel system which tends to focus on the higher level of only the most obviously important events. The significant events data base, I would say, covers the first 3 or 4 levels of significance.

The other data base that's become very effective is called the NPRDS data base, Nuclear Plant Reliability Data System. That database covers the behavior, the failure, lack of function, or degraded function of about 2000 components and systems. With about 30 systems in the PWR, about 20 systems in the BWR. It is now a very robust data base. One hundred percent of the plants in the US contribute to it. It's roughly 20 times as large, in terms of real data content, as was available at the time WASH-1400 risk study was performed in 1975. It's good enough that for many components you can do trend analysis, you can look at 3 or 6 month intervals and see if there is a trend one way or another and get some very important guidance from that kind of trend data.

Probabilistic analysis. Complementing the deterministic analysis, which is what we get heavily educated and trained and familiar with, is a newer discipline called probabilistic analysis. This has developed because people, good engineers, observed that sometimes when you do a calculation you can not make a single number as an input to the calculation. The numbers are inherently stochastic. For instance, if you do a fracture mechanics calculation and you want to know how many cycles to crack growth, it's a stochastic process, a statistical process. You really have a distribution function to cope with. Whenever you have a distribution function, whether you call it that or not, you're doing a probabilistic analysis. We now have a listing of about 27 completed and published PRA's and probably another dozen or so on the way. The methods and the data for performing probabilistic risk analysis are reasonably mature now. There's a handbook issued in this country and there is a committee in IAEA that's reviewing the same subject.

We held a long workshop several years ago of all the people that we could find who had actually performed PRA analysis over the previous 8 years or so and tried to get a definition of what leads to a very good and realistic probabilistic risk analysis. The nearly unanimous opinion was that you had to have plant people who had actually worked with the hardware and worked with the design to make sure that the models of the sequences that were put down on paper are physically realistic. You cannot do that from drawings. You almost have to walk in a real plant to be able to see the potential interactions in detail. If you don't do that, you get a generic PRA, which has some value but it doesn't give you as much insight to the operation, particularly to the potential hazards, as if you have that level of realistic detail. So EPRI conducted what was hoped to

DATA BACKUP FOR ENGINEERING ANALYSIS

- RELIABILITY DATA BASE AND PLANT EXPERIENCE BASE WERE SMALL, SCATTERED, LARGELY INACCESSIBLE AND UNANALYZED IN 1975.
- ENGINEERING JUDGMENT AND OPERATING EXPERIENCE WAS SCATTERED, FRAGMENTED AND LARGELY INACCESSIBLE.
- OPERATING EXPERIENCE DATA FROM ALL U.S. PLANTS IS NOW CONVENIENTLY ACCESSIBLE, INCLUDING ANALYSIS AND REMEDIAL ACTIONS TAKEN. (COVERAGE EXTENDS BACK TO 1980, AND INCLUDES SELECTED DATA FROM 13 OTHER COUNTRIES IN RECENT YEARS.)
- IAEA HAS ESTABLISHED A SIMILAR EXPERIENCE REPORTING SYSTEM (1985)
- COMPONENT RELIABILITY DATA BASE (NPRDS) IS ABOUT 20 TIMES LARGER THAN THAT AVAILABLE IN 1975, ALL U.S. PLANTS PARTICIPATE, TIMELINESS OF DATA GREATLY IMPROVED, TREND ANALYSIS AS WELL AS STATISTICS IS NOW POSSIBLE.

PROBABILISTIC RISK ANALYSIS - PRA

- PRA FOR 25 PLANTS COMPLETED - MORE UNDERWAY
- METHODS AND DATA MATURING
- A CRITICAL REQUIREMENT FOR GOOD PRA IS REALISTIC PHYSICAL INSIGHTS IN SETTING UP MODELS AND PERCEIVING POSSIBLE INTERACTIONS
- TEAM PERFORMING A PRA SHOULD HAVE ACCESS TO EXPERIENCED DESIGNERS, DEVELOPERS AND OPERATORS TO SUPPORT PRA SKILLS
- PRA ON A GENERALLY SIMILAR PLANT IS USEFUL FOR DESIGN REVIEW AND AS A GENERAL GUIDE FOR SAFETY MANAGEMENT, BUT PLANT-SPECIFIC PRA IS PREFERABLE

be a kind of a model PRA with very large resources and a very mature team. I think we had something like 6 operators assigned from the operating company to work with the statisticians and the engineers and the analysts in making sure of the physical models of how the different systems work.

As you know, a good PRA has roughly 100,000 of these line diagrams that describe possible sequences. Even at that level, there may be some interactions that have been overlooked. If you have not been involved in PRA, probabilistic risk analysis, its easy to be overwhelmend with the large mass of mathematics that's used. But the mathematics is basically very simple. It boils down to Bayes theorem, which you can learn in 15 minutes. It's a generalization of Bayes theorem to combine the probabilities of things that might happen simultaneously, or that are in parallel. Then people have invented some very sophisticated mathematics to make the handling of this tremendous mass of numbers more convenient. But the essential thing is to get a model of the particular event sequence correct.

Sequence risk analysis. For much of what I will talk about on the decision making in plant management, the important thing is a little piece of the PRA; its a Sequence Risk Analysis. Say I have a particular valve that is sticking sometime or a particular relay that does not have as high a reliability as I'd like and then I have to make some decisions. Do I shut down? Do I replace? Do I maintain it more often? I really want to know what are the kinds of events that that particular deficiency can affect and what consequences do they have. If I'm interested in a little microscopic piece of the overall PRA, that's not very different from what a good designer, operator, or manager would have done anyhow without PRA. Intuitively you get back to the right question. What are the things that could go wrong if this thing is not functioning properly. Then you make the decisions on how important it is and what your options are for fixing it.

Even though there are 27 PRA's now, there are over 200 plants operating. So there is roughly 6 times as many plants that don't have detailed PRA's as there are that do have. So how do you get nourishment from this discipline? Well, there are two ways. If you have a particular event or deficiency of concern or improvement that is offered, you can ask what other plants have a system similar to mine that involves that improvement and look at the PRA for the plant that has that similar system. Even is there are subtle design diffe ences, you can get a great deal of nourishment from looking at somebody else's PRA and saying, "For that sequence, what is the effect on my plant?" If it's not exactly the same design, you can sometimes make adjustments for that. So the plant-specific sequence analysis is always available. You can look at a PRA for a similar plant and you say this is a thing I'm worried about and then you can do a local model, a local sequence of the item of concern. You can model that locally and do what then amounts to a deterministic analysis using only a few probabilistic numbers to help make the decision for replacement or rework or whether to implement a proposed improvement. It's also sometimes helpful if the regulatory authority thinks you should make a change and you can say, "But I already have great safety on that particular threat." You might even make that case also that a proposed change is counterproductive because it raises the chances of another damaging sequence.

What are the more general uses of the PRA? What I've talked about so far are the uses which bear on the ordinary difficulties in the plant, either in design, construction or operation. No matter what you build, it will have some deficiencies. In fact, if you look very hard you can find literally hundreds of things which at that moment are not in the perfectly ideal condition. And the ideal of the plant management is to make that cloud of not-quite-perfect things as small as possible. I have a theorem, which I can't prove; it's a conjecture, that the probability of really serious events goes something greater than the second power as a function of the population of the minor deficiences that are present in a plant at a given time. So good management from an overall safety standpoint comes to the same point of view as reliability to get good productivity. The two motives go together.

Another function of PRA is that it is the only logic tool you have to know how well off you are with respect to low-probability, high-consequence events. There is no other way to handle that question. You can use engineering judgment and people complain that the uncertainty of a PRA

USES OF PRA

- HIGHLIGHTS THE SEQUENCES, SYSTEMS AND COMPONENTS THAT CONTRIBUTE MOST TO RISK

- "OUTLIER" SEQUENCES PROVIDE OBVIOUS TARGETS FOR RISK MANAGEMENT EVALUATIONS AND ACTIONS TO REDUCE RISKS

- FOR LOW PROBABILITY, HIGH-CONSEQUENCE ACCIDENTS, PRA PROVIDES THE ONLY SYSTEMATIC MEASURE OF RISK IMPORTANCE AND OF THE RISK REDUCTION EFFECTS OF REMEDIES

- FOR MORE FREQUENT EVENTS OR DEFICIENCIES, PRA CAN BE USED IN THE PLANT TO PRIORITIZE IMPROVEMENTS NEEDED, AND TO EVALUATE REMEDIES

- PRA IDEALLY IS A LIVING DOCUMENT AND PLANT MANAGEMENT TOOL, UPDATED AS EXPERIENCE ACCUMULATES, AND AS CHANGES ARE MADE IN SYSTEMS OR COMPONENTS is a factor of 10 or 100 log 1 or 2 in either direction, but if you do it by the seat of the pants engineering judgment, the uncertainty is even much greater. So it's the only way to systematically take a look at what are the threats to the plant and what are the remedies I have in place to cope with those threats. It has been the most basic tool to change our understanding of the low probability and high consequence events. Many other industries are also finding that they must learn the same discipline of looking at highly improbable events with large consequences by a probabilistic technique because there is no other way. You can do a very useful thing intuitively, that is if you say I will make my plant extremely reliable and do the maintenance very promptly and have very well trained operators, intuitively you suspect that you have done a good thing with respect to low probability high consequence accidents. However, you can't prove it and you can't have a feeling for how much benefit you get for a given expenditure of effort. The PRA then gives you another thing, it gives you a tool for measuring how big an improvement you get for a given change or a given improvement in training or a given improvement in equipment. The relative measure of probability due to a change is much less uncertain than the absolute values.

Limitations of PRA. Now, having said all that, let me say what PRA cannot do. Like any new discipline, some practioners may say it's the magic pill to cure all ills - it definitely is not. The limitations are listed here. It obviously doesn't prove that the plant is safe enough. PRA can show very good numbers and if I go in and disable a few relays tomorrow it's not a very good plant anymore. The PRA only tells you in principle that, given that you have good managment and operation, you have a fine piece of equipment, a fine system but you can't use it as a proof that it is safe enough. Secondly, the uncertainties in modeling as I mentioned, both the uncertainties in the combination of the statistical data and the uncertainties in the modeling, even the optimist will admit give factors of 5 or 10 uncertainty in the mean value. If somebody asks the 95% confidence limits, you often will get log 2 as a plus or minus on the number. Therefore, it's very important to get a good perspective on the uncertainty. Now with a tool that is so uncertain, you can ask yourself why is it any use at all? There is a secret weapon hidden in this

LIMITATIONS OF PRA

- DOES NOT ASSURE THAT ANY SPECIFIC PLANT IS "SAFE ENOUGH"

- UNCERTAINTIES IN MODELLING, IN GENERAL STATISTICS AND IN DIFFERENCES WITH LOCAL STATISTICS RESULT IN 5X TO 10X UNCERTAINTY IN MEAN VALUES OF PROBABILITY, 95% CONFIDENCE LIMITS EVEN WIDER

- A PRA IS SNAPSHOT; IT SHOULD NOT BE REGARDED AS PREDICTIVE OVER LONG PERIODS OF TIME BECAUSE:

  - WIDE UNCERTAINTY LIMITS

  - CHANGES OVER TIME BASED ON LEARNING FROM EXPERIENCE

  - LOCAL HIGH LEVELS OF MANAGEMENT MAINTENANCE AND OPERATING KNOWLEDGE & DISCIPLINE

  - LOCAL LAPSES IN MANAGEMENT, MAINTENANCE OR OPERATION uncertainty which is very important to recognize and, unfortunately, its not yet being sufficiently used this way. The secret weapon is that the probabilities of a particular sequence, the sequence risk analysis, (1) as it is now, (2) or as it is with a deficiency, or (3) as it will be with a given improvement, the ratio of those three numbers is known with great accuracy. This is true because most of the uncertainties cancel out. As a decision-making tool, it's a very powerful tool, both for regulation and for actual plant management. It is being used that way to a very considerable extent in plant operation in the United States. Regrettably, as we know from the status of nuclear energy here and the delays on the plants in this country, this discipline has been less common on the construction side. So that any deficiency no matter how small (even if there isn't a proper signature on a piece of paper or radiograph) can delay the project. In many - perhaps most cases, that's a failure to recognize the relative importance of different deficiencies.

This relative importance measure is extremely valuable, but because it is valid only as a snapshot in time, it's a useful guide only if you maintain the right operating culture. That means that it is an equal responsibility of the regulatory and of the operator. If the operating culture changes, all the numbers will change.

<u>Reliability through systematic learning from experience</u>. There's another related aspect, namely, that there is learning from experience. Those of you who have had industrial engineering or reliability engineering know there's something called the Duane reliability curve, that is, if you take almost any heavy piece of equipment and measure the mean time to major repair or failure versus total experience, you get roughly a straight line on a log-log plot. And the time to major overhaul increases as the millions of hours of operating experience increase. This relationship holds <u>only in the situation where you have a continuing process of record keeping of deficiencies and keep improving on that experience</u>. In the nuclear business, this has existed systematically and comprehensively in this country only for about 5 years; in France, for about 3 years, and starting in IAEA for about 1 year. Of course feedback from experience has occurred ever since the start of nuclear power, but until recently it has

been parochial, fragmented, and not generally available to all who need it. Feedback from all experience, at a detailed engineering level, is still a relatively new process, but it is happening. If you have a component failure, usually it's because some part failed or something was vulnerable to the environment or dirt. If you have a systematic record-keeping and feedback system, then the reliability, mean time to failure or to major overhaul goes up with time.

Let me summarize the subject of PRA with just two points. By the use of the relative risk assessment it gives you a very powerful management tool for proper allocation of resources. (If you over-allocate resources to a less important problem, you are making the system less safe.) It is not necessarily noble to overreact to a small issue. But you have to know which is a small issue and which is a big one. The PRA gives you one of the most rational tools to determine that. Secondly, in terms of public risk, Western world power reactors have an extremely good record. We haven't hurt anybody as near as we can tell. However, we have damaged equipment and we have lost one plant very visibly and several plants less dramatically, where they had some deficiencies that were too expensive to repair or redesign so it was cheaper to decommission the plants. The use of the structured decision-making system can give you the confidence that your particular situation is several times safer than experience history so far. I would say that right now, on the average in this country, plants are about 5 times safer (in a certain, carefully defined way) than they were in 1980, because we have made a number of changes. So the perception, both in the probability of a public-affecting accident and the consequences of it, is much improved over 1975. The real issue now, (and Dr. Starr who is lecturing later, made this point intuitively a good many years ago), is that the risk to the plant measured in any terms, human or financial, is at least 50 times greater than the risk to the public. So your motivation is to protect your plant. You have to keep your plant healthy because there is a chance you can lose your plant.

We are talking to this group as decision makers or potential decision makers on very important issues on either regulation or construction or operation. One of the things that gives you some comfort is that we have

THE BASIC VALUES OF SYSTEMATIC USE OF PRA

I. PROVIDES MEANS TO GUIDE THE BEST USE OF RESOURCES; BY PROVIDING CONTINUING MEASURES OF THE RELATIVE SAFETY BENEFITS; SO THAT EFFORTS AND RESOURCES ARE SCALED TO THE RISK REDUCTIONS ATTAINABLE

II. CAN PROVIDE ASSURANCE THAT THE SPECIFIC, LOCAL RISK EXPOSURE LEVELS ARE SEVERAL TIMES LOWER THAN HISTORICAL EXPERIENCE

- TO THE EXTENT THAT CONTINUED AND DISCIPLINED APPLICATION OF THE LESSONS LEARNED FROM CUMULATIVE WORLD EXPERIENCE, GUIDED BY PRA FOR ESTIMATES OF IMPORTANCE, IS MAINTAINED BY MANAGEMENT AND OPERATING STAFF the kind of experience which is codified in codes and standards and, ideally, also in good regulations. Increasingly, there are also tabulations of what's called good practices. People have written books and topical reports and symposia on good construction practices and good design practices and, increasingly, on good operating practices. INPO, as a part of its plant visits, regularly puts out a list of things which this plant does especially well, which other people don't do as well, for the benefit of others. Again, that is a cumulative learning experience. These are very good general guides for how to do well, but they are not very specific and they are open always to interpretation. That's where skill and experience comes in.

Optimum management of risk exposures. We can define intuitively what the optimum risk management decision process should include. (We're trying to do it in a more scholarly way in some papers that are coming up next year.) Resources are always finite so you must always have a good set of algorithms for how you allocate resources from the less to the more important things. The ability to discriminate high, medium, low and negligible or de minimis levels is a very basic skill. Ideally that is done with good communication between the operator and the regulator because if they have different opinions on what's important you get a chaotic situation.

The question of timing of resources is also very important. If you ask Mr. Bernero when he talks, he'll give you his rule-of-thumb, which is that if you have a measure of the contingent probability of a given deficiency it gives you a feeling for how quickly you should correct that deficiency. In round numbers, if it's a contingent probability of high 10^{-3} merge (probability of severe accident per year) you fix it right away. If that means shutting down the plant or stopping the construction, so be it. If it's in the 10^{-3}-10^{-4} range, you can take varying lengths of time, months, years. Below 10^{-4}, you can allocate some resources and do it when you can, but not necessarily with deadlines. The French have implemented most of the backfits and safety improvements that the U.S. program has considered and they've taken very little penalty in construction or operation schedule. I think the main difference is that they have generally made that discrimination of the 10^{-4} and lower deficiencies. For

ROLES OF CODES, STANDARDS, AND REGULATIONS

GENERAL CRITERIA	LOCAL EXPERIENCE { UTILITY, SUPPLIERS, CONSULTANTS } ENGINEERING JUDGMENT APPLICABLE CODES AND STANDARDS ANSI, IEEE, ASME, ETC.
REGULATIONS	RULES GUIDES BULLETINS ORDERS LICENSE REQUIREMENTS TECHNICAL SPECIFICATIONS
GOOD PRACTICES	SYSTEMATIC TRACKING OF RESULTS REMEDIES FROM EVENTS ANALYSES AND FAILURE ANALYSES INPO COMPILATIONS - OPERATIONS & MAINTENANCE SUPPLIER COMPILATIONS, CUMULATIVE, SYSTEMATIC STAFF EXPERIENCE

THESE INPUTS

A. CAN DEFINE OR LIMIT OPTIONS FOR REMEDIES
B. CAN BE USED TO MINIMIZE ANALYSIS
C. DO NOT INSURE SUCCESS OF REMEDY
D. OFTEN OPEN TO A RANGE OF INTERPRETATIONS
E. SHOULD NOT BE RELIED ON TO AVOID DETAILED ANALYSIS AND/OR TESTING IF THE SPECIFIC APPLICATION HAS SOME NOVEL ELEMENTS.

OPTIMUM RISK MANAGEMENT DECISION PROCESSES

- GOAL IS TO PROVIDE THE BEST ATTAINABLE LEVELS OF SAFETY AND PERFORMANCE WITHIN NORMAL LEVELS OF RESOURCES

- DEFICIENCIES IN DESIGN, CONSTRUCTION, OPERATION, OR MAINTENANCE CAN BE CLASSIFIED BY THE ESTIMATED LEVELS OF SAFETY IMPACTS

  HIGH

  MEDIUM

  LOW

  NEGLIGIBLE

- THE TIME ALLOWED TO BRING IN REMEDIES SHOULD BE ROUGHLY THE INVERSE OF THE SAFETY IMPACT (EXPECTED VALUE) such problems, it is practical to take until a second or third refueling to fix. With this decision made there are relatively few items put on critical path. That same discrimination is also growing more in this country but is sometimes overlayed by procedural matters. This is the same chart but with some of these numerical criteria offered very tentatively. Other people would argue for somewhat different numbers but at least these are in the general range.

I would need much more time to define contingent probability closely. For those of you for whom that is a familiar concept and without any theoretical basis, the practical observation worldwide is that contingent probabilities in the range of 10^{-5} are do-nothing level. People have generally decided that a probability of about that order is not worth fixing and/or there are usually more important items still pending action. That's a de minimis decision; it's very important to have such a level in mind because if you chase everything you will necessarily miss some of the more important ones. Some sort of a rationale of this kind is essential because those kind of decisions in most industry are made implicitly. You don't write down the numbers, you don't face those risks explicitly, you make them by experience and judgment and by actual experience of damage and ----. You make some aircraft decisions by how many crashes there have been. It's not practical to have crashes to learn about nuclear power plants. We have to try to learn in less expensive ways.

Here is an area where the motivation and the needs of both the regulator and the builder and operator should be nearly identical. There's the management role to see that resources are effectively used. Both operating management and the regulatory management have the obligation to track what is done. Are the things being done that are being directed? Are the things being done as planned and scheduled? Are other things happening which might defeat the effectiveness of improvements? Say an improvement is agreed on and scheduled, the money is spent, and somehow you go and look in the plant and its not working. When you realize the thousands and tens of thousands of items that must be tracked, you need to have an army of clerks and a warehouse full of filing cabinets and a lot of people running around. The more realistic way now is to have a good computer tracking system plus enough trackers to keep the picture up to date.

OPTIMUM RISK MANAGEMENT DECISION PROCESSES

- GOAL IS TO PROVIDE THE BEST ATTAINABLE LEVELS OF SAFETY AND PERFORMANCE WITHIN NORMAL LEVELS OF RESOURCES

- DEFICIENCIES IN DESIGN, CONSTRUCTION, OPERATION, OR MAINTENANCE CAN BE CLASSIFIED BY THE ESTIMATED LEVELS OF SAFETY IMPACTS

	CONTINGENT PROBABILITY
HIGH	OVER 10^{-3}
MEDIUM	$<10^{-3}$ TO $2x10^{-4}$
LOW	$2x10^{-4}$ TO $2x10^{-5}$
NEGLIGIBLE	BELOW $2x10^{-5}$

- THE TIME ALLOWED TO BRING IN REMEDIES SHOULD BE ROUGHLY THE INVERSE OF THE SAFETY IMPACT (EXPECTED VALUE)

INTEGRATION AND TRACKING OF RESOURCES

CRUCIAL MANAGEMENT ROLES:

- **MAINTAIN ADEQUATE LEVELS OF RESOURCES:**

  PERSONNEL
  SPECIAL SKILLS
  SUPPLIES
  TECHNICAL SUPPORT
  FUNDING
  SUPERVISION

  GUIDELINES FROM:
  TYPICAL INDUSTRY LEVELS
  EVENTS & TRENDS

- **TRACK THE EFFECTIVE USE OF RESOURCES**

  MONITOR CONSTRUCTION OR REWORK
  MAINTENANCE
  OPERATION
  PROCUREMENT
  CONTRACTORS

  INDICATORS

  COMPARISONS
  TRENDS

- **PRIORITIES AND SCHEDULES**

  RELATED TO ESTIMATED RISK EXPOSURE REDUCTIONS EXPECTED

Basic responsibilities of management. My closing chart lists the issues faced by those of you who have actually operated or had responsibility for a big operation. At some point you have a problem that you feel very strongly about. It's a gut problem, and these are some of the gut problems. How safe is safe enough? We are trying to invent numerical criteria for that, but it's really a much more complicated problem because there are many issues that are not treated by numerical standards as to how safe is safe enough. How fast should I get the improvement or the change or the review or the analysis done? Should I put in a temporary patch or should I go for a good long term design fix? And finally, if I am putting in a long-term fix, what should I do to notify people in the meantime to be alert, to be more careful, to inspect more closely, and to cope with the potential problems? So these are the heart-rending, soul-wrenching decisions that people who have the responsibility must face. The various tools and techniques I've mentioned are important inputs to making such decisions, but ultimately it requires the skill and judgment and experience of people to make these decisions on a rational and defensible basis.

BASIC MANAGEMENT RESPONSIBILITIES INVOLVING JUDGEMENTS

CRITERIA: "HOW SAFE
IS SAFE ENOUGH?

HOW SOON SHOULD A
REMEDY BE REQUIRED?

LONG-TERM V. SHORT TERM
REMEDIES, OR BOTH?

INTERIM MEASURES?
ALERTING, TRAINING, PROCEDURES
ADDED SURVEILLANCE

X MANAGEMENT SIGNIFICANCE OF RISK ANALYSIS

Chauncey Starr
Vice Chairman
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MANAGEMENT SIGNIFICANCE OF RISK ANALYSIS

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INTRODUCTION

The contribution to the reduction of public risks by modern risk analysis, involving quantification of system event probabilities and their consequences, is best understood by considering the accepted approaches to risk prior to the middle of this century. Civil engineering structures -- buildings, bridges, dams, etc. -- are classic historical examples. The historical design objective was to avoid failure of the structure, defined as collapse under expected usage. To provide such assurance, the designers applied a traditional "safety factor." For example, if a rope was tested to hold 100 pounds, a safety factor of 10 would be provided if the maximum load did not exceed 10 pounds. In practice, these safety factors traditionally ranged from a low of about 3 to as much as 40, depending on the designers' judgment and the tradition for each type of usage, i.e., steady state, cyclic stress, shock, corrosion, etc. Thus, the safety factor supplied a design umbrella large enough to cover all the areas of the designers' known range of ignorance, i.e., the "known unknowns." The system worked reasonably well, although an occasional structure collapsed because of an "unknown unknown"; for example, the Tacoma bridge collapse caused by unanticipated wind-induced oscillations.

The safety factor design approach was socially acceptable at the time. The engineering profession said, "trust us," and the public did. There were no probabilistic risk assessments involving off-design failure analyses, no environmental impact statements, nor any of the other modern trappings of project reviews. The designers' judgment on the choice of safety factors integrated all uncertainties without an explicit justification of the choices. The public risk was implicitly covered by the design objective of avoiding failure, but was never explicitly estimated. When the unforeseen occasionally occurred, it was usually accepted as an "act of God."

The historical approach to the risk management of a replaceable product which permitted experience feedback was one of empirical "trial and error," as, for example, with autos and airplanes. Operating experience was fed back to guide improvements, a process that continues today. The traditional "safety factor" was less important in such product designs, because the feedback process was sufficiently rapid (a few years) to permit improvements needed for achieving a performance target. The collective risk was initially low, because only a few individuals were involved in the early developmental stages, although individual risks were high.

It should be recognized that the "safety factor" and "trial and error" methodologies continue to be pragmatically useful, and are only slowly being replaced by modern risk assessment approaches in a limited number of publicly pervasive systems. The penetration of large-scale technologies has become much more rapid than decades ago, so the "trial and error" method can be very costly both in public health and cost. Further, some large-scale systems involve so many interdependent components, that the individual "safety factor" approach would be compounded to the point of making the system inoperable (e.g., air transport). Finally, very rare but high consequence events may require decades or centuries to provide the feedback information for guiding decisions, and each such occurrence may be undesirably costly to public health and safety. It is these considerations that have encouraged the development of modern risk assessment approaches.

MODERN RISK ANALYSIS

The objective of modern public risk assessment is to provide a basis for actions to minimize the impairment of public health and safety arising from technical systems creating risks. This objective is not directly concerned with the ability of a technical system to perform its functions, in contrast to the historical "safety factor" or "trial and error" approaches. The risk assessment focus is on injury to the user and the public, with the technical equipment being considered a potential risk-creating source.

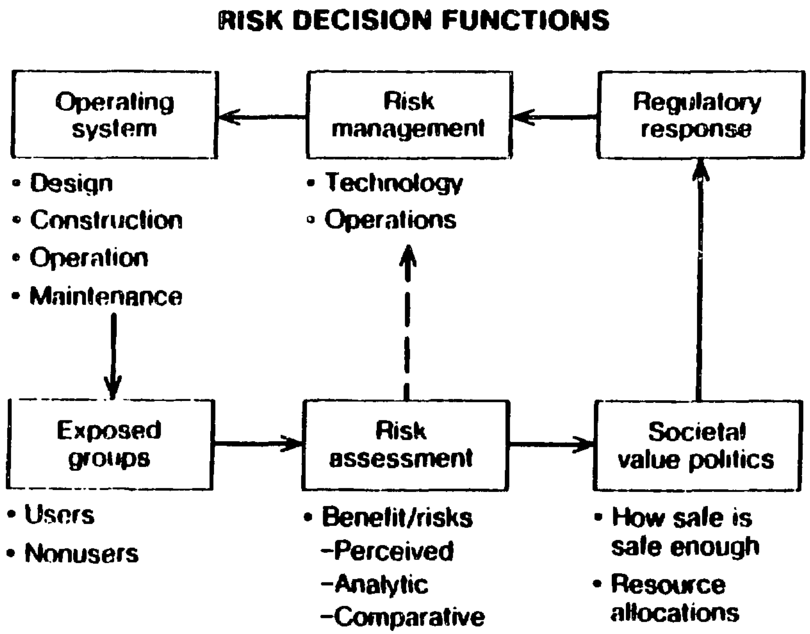
The analytical approach to estimating the probability of each event in such a system analysis utilizes empirical data when it is available, or experience with similar circumstances and equipment, or professional judgment based on a composite of experience. The same situation applies to consequence estimates. Thus, risk analysis generally embodies the heuristic approach of empirical learning, with large uncertainties in event probabilities and public consequences. Nevertheless, the central values of the final estimates do provide a "best knowledge" estimate of the relative importance of a risk. Further, the detailed analysis of the off-design and failure modes of the system provide a very useful disclosure of the key components or subsystems which have most influence on the public risk. Such insight guides redesign and operating and maintenance techniqeus tailored to reduce risk.

A noteworthy example is the recent use of Probabilistic Risk Assessment (PRA) for nuclear power plants. These have not only provided a better professional estimate of failure probabilities, but also have stimulated technical fixes and wiser off-design operational responses. Such studies also provide greater confidence to operators who must act in emergencies, because they understand more completely how the system will respond to their measures. The nuclear utilities are now undertaking PRAs voluntarily in recognition of these benefits.

RISK ANALYSIS AND MANAGEMENT

As shown in Figure 1, the overall process of society's approach to risks involves a sequence of steps, each requiring an action by a group with a delegated responsibility. All societies, regardless of political organization, involve such decision steps. The issue is always to whom are the respective responsibilities for each step delegated and on what information they act.

The situation is substantially different when we address the societal question of "how safe is safe enough." The implicit end-result of a society's answer to that question is the allocation of the resources needed to achieve an agreed-upon safety goal. Unfortunately, in all societies such an allocation is part of a "zero-sum" game, i.e., resources



RISK DECISION FUNCTIONS
Operating system
Risk management
Regulatory response
• Design
• Construction
• Operation
• Maintenance
• Technology
• Operations
Exposed groups
Risk assessment
Societal value politics
• Users
• Nonusers
• Benefit/risks
-Perceived
-Analytic
-Comparative
• How safe is safe enough
• Resource allocations

Fig. 1

applied to one goal leave less available for other goals. Thus, this competition among social goals inevitably involves every special and group interest that influences a society's decisions.

It is interesting to observe that the setting of safety goals, either absolute or comparative, provokes intense debate both professionally and publicly. Nevertheless, such goals are usually secondary in practical importance to the many implicit and obscure decisions which allocate the resources to achieve the goals. Safety goals have great political currency, since they embody idealistic consensus views on health and safety. I doubt if anyone wants to be exposed to toxic substances, or to die by accident or from disease. Politicians do not get elected by voting for exposure to carcinogens. But, clearly, we are not spending unlimited sums to achieve health and safety goals. Every society has many social goals, and the competition among these limits the allocation of resources the health and safety. As a result, every society determines an acceptable "non-zero" risk level for each of its activities.

The factors entering into a determination of an acceptable risk level broadly involve the societal benefits and costs and the available resources. These factors include both tangible and intangible aspects, and are weighted by social values and public perceptions. The common bureaucratic phrase, "benefit/cost ratio," sometimes applied to evaluations of risk management, is overly simplistic, and is useful only for narrow issues involving small costs. The broad social objective of risk analysis is, therefore, the most effective use of the resources allocated to public health and safety.

RISK MANAGEMENT

Risk management is finally carried out always by individuals, companies, or other operating units -- not by regulating agencies. The function of regulatory agencies concerned with public health and safety is to assure that risk management techniques are implemented in operating systems which involve the public, as shown in Figure 1. I will not here discuss the organization and operations of regulatory agencies, a much belabored subject. I will address some of the policy issues involved in risk management which may determine its effectiveness.

In principle, the objective of risk management of a specific activity is to minimize social losses arising from an existing or potential risk. The preceding political process in Figure 1 presumably should have considered the issues of societal benefits and national resources, and should have defined for the regulatory agency the criteria for imposing remedial costs upon society. In practice, an image-motivated political body may vaguely direct a regulatory body to minimize both the risk and social cost of doing so, thus transferring to the regulatory body the political chore of balancing societal benefits, costs, and resources under a mandate that the public be protected from unreasonable risks.

The regulatory techniques of risk management fall into two classes: (1) imposition of technical and operating criteria; and (2) encouragement of operating system self-management. In both cases rewards and penalties are used to enforce these objectives. Rewards include licensing (or the equivalent approval to sell) and support of public acceptability. Penalties include a range of punitive actions, liabilities, and, most importantly, a degradation of public acceptability. The effectiveness of these regulatory techniques has been much studied, debated, and reviewed. I will not discuss them further. However, it is useful to consider the basic limitations common to all such regulatory actions.

The effectiveness of risk management is constrained by the complexity of most risk situations and their uncontrollable factors. While frequently occurring risks (e.g., auto collisions) provide an empirical base for determining many of the parameters involved, this is not the case for rare occurrences or for statistically low-level risks obscured in a large aggregation of similar consequence events. Thus, the predictability of the outcome of a risk management action is often severely limited. Because most such actions involve significant resource costs, their unpredictable outcome tends to discourage all but the most obvious measures.

The infrequent but high-consequence events present special problems of predictability and risk management. Every accident is the end result of a chain of events starting with some small initiator. There are usually a very large number of such potential initiators, each starting a different chain of events. For high-frequncey risks, the empirical data base usually discloses the most common consequence, and the thus provides a useful risk management opportunity. For infrequent accidents, a very few sequences may have been observed, but managing these provides very little assurance that the potential spectrum of initiators and sequences has been importantly reduced.

The most extreme risk scenarios are predominantly based on hypothetical rare sequences (e.g., the risks of nuclear power). This leads to risk management approaches which concentrate on virtuous "good practices," such as frequent maintenance, component testing, meticulous supervision, operator training quality, sobriety, alertness, honesty, cleanliness, etc. Of course, technical modifications to existing system are included in risk management actions to address perceived defects, but it is often controversial that they actually reduce potential risks. The basic difficulty with rare event risk management is that the paucity of empirical information forces a dependence on unverifiable professional judgment in fields of great uncertainty. This is also the case for very low-level effects. Thus, public anxiety cannot be allayed for visible proof, and may, in fact, be enhanced by visible risk management. Such anxiety leads to a continually increasing political demand for further risk reductions, continuing public anxiety and social expenditures disproportionate to the real risk.

PERCEPTIONS OF NUCLEAR RISK

One can easily point to a variety of different perceptions of nuclear risks. Nuclear proponents and critics disagree about the magnitude and even the nature of the risks, within very broad limits set by operating experience. Most nuclear professionals believe that reactor safety has now reached reasonably acceptable levels. Given the inevitable absence of sufficient empirical information on such low-frequency events, and therefore on the consequences of these risks, the controversy will continue for a long time.

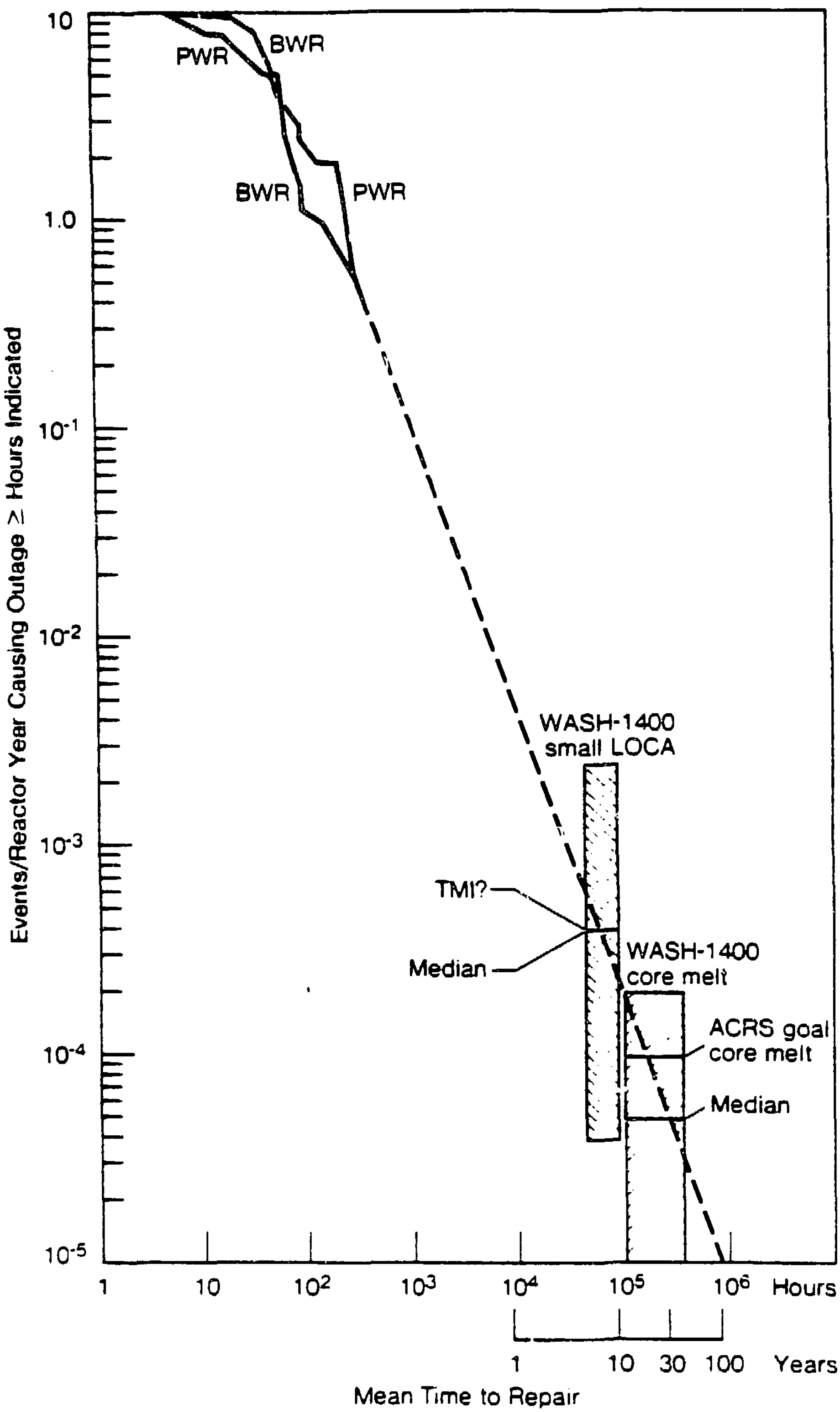
There is one commonly held perception about nuclear power plant risk that bears further scrutiny. This perception is that without strong regulations and oversight by government, nuclear power would be much more hazardous than it currently is. A contrasting viewpoint has been suggested, namely, that the possibility of large financial losses from a nuclear accident provides incentives which are sufficiently strong to lead a utility to build and operate plants with very low public risks. This idea has not been generally accepted. At a time when the political popularity for using self-interest or economic incentives as regulatory tools has been rising, the approach to nuclear safety regulation has continued in the direction of strict rule, policing, and penalty.

The issue of the strength and nature of the safety incentives provided by the risk of financial loss is amenable to analysis, and is central to a discussion of the possibility of using utility self-interest to provide safe plants. As a starting point, the Three Mile Island accident can be considered. Who lost what due to the accident? The most significant public consequences were principally due to the anxiety suffered by nearby residents; it is difficult to assign a value to these costs. Other public consequences include expenses for temporary relocation and business disruption. The utility which owns TMI suffered a loss which may reach a billion dollars. Bankruptcy has been averted so far, but barely. The long-run assignment of losses to utility stockholders, bondholders, and ratepayers is not yet predictable. While one data point, such as the TMI accident, does not make the case, it is illustrative that in this worst accident to date, the utility was the biggest loser.

The following analysis estimates the distribution of public and utility risk in general. These results indicate that potential utility losses are several magnitudes greater than potential public losses.

COMPARING PUBLIC AND UTILITY RISKS FROM NUCLEAR POWER

A frequency-sequence estimate for nuclear power plant accident and equipment failures was obtained by interpolation between data from operating experience and the estimate of the probability of a core melt from WASH-1400, as indicated in Figure 2. The upper portion of this curve was derived from data collected for EPRI.



Events/Reactor Year Causing Outage ≥ Hours Indicated
10
1.0
10-1
10-2
10-3
10-4
10-5
PWR
BWR
BWR
PWR
WASH-1400
small LOCA
TMI?
Median
WASH-1400
core melt
ACRS goal
core melt
Median
1
10
10²
10³
10⁴
10⁵
10⁶
Hours
1
10
30
100
Years
Mean Time to Repair

Fig. 2

The second component of this calculation is the assignment of costs to outages of various durations. Figure 3 indicates the estimated outage cost for nuclear power plants. Outage costs were assumd to be $1,000,000 per day for outages of a few years or less. Longer outages (including those resulting from accidents which would ruin a plant) have decreasing costs per unit of outage, with the maximum loss set at roughly $2.4 billion.

Public risks are compared to utility financial risks in Figure 4. The utility loss curves were derived from the curves of Figures 2 and 3, and the effects of insurance are included. The resulting loss curves are indicated in Figure 4. The expected costs of the frequency-severity curves in Figure 4 are indicated in Table 1.

COMMENTS ON THE ANALYSIS

As Figure 4 and Table 1 indicate, the median estimates of the expected public risk costs is only about one-fiftieth of the utility risk costs. This disparity is fairly insensitive to alternative value sets and to different costs and risk estimates. It is particularly insensitive to the social cost assigned to a fatality, given that the ratio of utility risk to the social cost of early fatalities is calculated to be roughly 10^4:1. Even if a higher value were used for early fatalities -- for example, if aversion to multifatality accidents were assigned a high value -- it is unlikely that the overall finding would be altered.

IMPLICATIONS FOR THE UTILITY INDUSTRY

From the viewpoint of a utility operating a nuclear power plant, this analysis indicates that safety requirements established by NRC to manage public risks do not provide an adequate level of financial protection to the utility. It is clear that many nuclear utility managers recognize that this is true, particularly since the TMI accident. The industry subsequently created the Nuclear Safety Analysis Center (NSAC), the Institute for Nuclear Power Operations (INPO), and the Nuclear Electric Insurance, Ltd. (NEIL) to help prevent or offset financial losses arising from nuclear accidents.

Cost Versus Outage Time

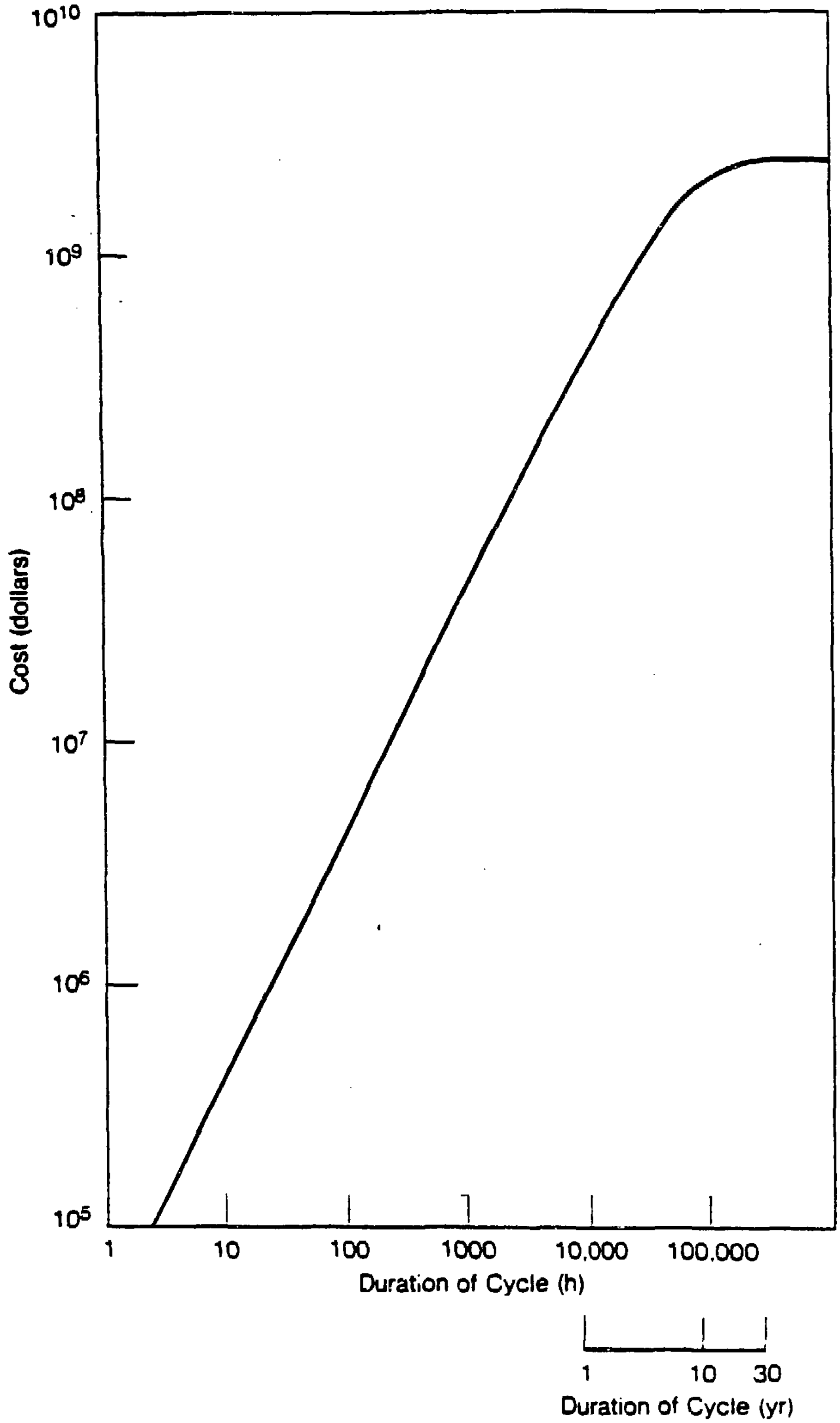


Fig. 3

Probability Distributions for Costs Arising From Reactor Accidents and Outages

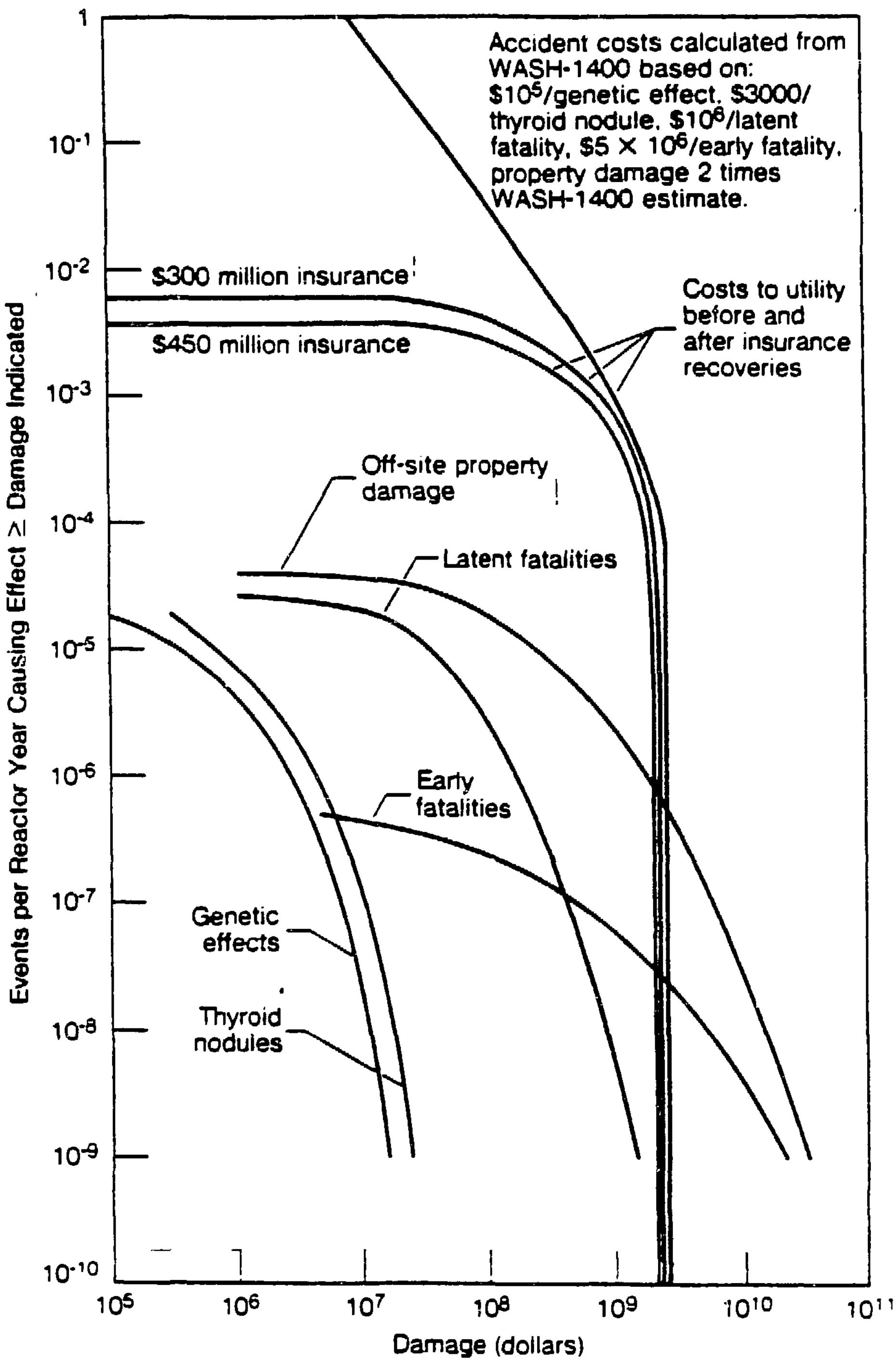


Fig. 4

Table 1

Public Risks-Expected Value Per Reactor-Year

Effect	Expectation*	Value	Expected Cost
Early fatalities	3×10^{-5}	$\$5 \times 10^{6}$	$ 150
Early illness	2×10^{-3}	$\$10^{4}$	20
Latent fatalities	7×10^{-4}	$\$10^{6}$	700
Thyroid nodules	7×10^{-3}	$\$3 \times 10^{3}$	20
Genetic effects	1×10^{-4}	$\$10^{5}$	10
Property Damage	$20,000	Twice WASH 1400	40,000

*Source: WASH-1400, Table 5 - 6

Utility Risks-Expected Value Per Reactor-Year

With $450 million insurance	$\$2.1 \times 10^{6}$
With $300 million insurance	2.9×10^{6}
No insurance (includes accidents causing 10 days outage or longer)	$\$24 \times 10^{6}$

IMPLICATIONS FOR REGULATORY RELATIONS

The usual basis for government regulation of risk is that a lack of sufficiently strong incentives for a risk producer to self-regulate leads to unacceptable levels of public risk. As the above analysis has indicated, the incentives to operate nuclear plants safely are extremely strong. Nuclear utilities are particularly suitable for safety motivation by financial self-interest. It is evident that nuclear accidents that cause large internal financial losses are more probable than those that might harm the public. Further, the rating of the utilities with their financial creditors would be considerably enhanced by expectation of reliable and safe operation. And, of course, the public acceptance of nuclear utilities as beneficial institutions would also be improved.

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XI INTERNATIONAL AND DOMESTIC INSTITUTIONAL ISSUES FOR PEACEFUL NUCLEAR USES

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INTRODUCTION

At the inception of peaceful nuclear uses, the international community found traditional institutions lacking and forged a relatively unique structure consisting of an intricate array of international and bilateral treaties, agreements, and exchanges.

International nuclear commerce for peaceful purposes began with U. S. President Dwight Eisenhower's "Atoms for Peace" speech before the United Nations on December 8, 1953. In this address, President Eisenhower called upon nations to engage in international cooperation for peaceful applications. To facilitate this effort, he proposed the establishment of an International Atomic Energy Agency under the aegis of the United Nations for distributing and safeguarding nuclear material and equipment. In this way, it was felt that the international community could ensure the orderly development of nuclear energy for peaceful purposes.

The idea of an international nuclear safety training program was proposed by the Department of State in a speech by Assistant Secretary James L. Malone in January, 1983. This concept is a further step in the continued cooperation among nations on peaceful nuclear uses that has expanded and matured through the years.

GOVERNMENT-TO-GOVERNMENT AGREEMENTS

Prior to the establishment of the IAEA several events occurred which had a substantial impact upon later developments. Within a year after President Eisenhower's address, the United States enacted the Atomic Energy Act of 1954 which authorized the U.S. to engage in nuclear trade with other nations where the recipient formally pledged not to use the assistance for military purposes. As a condition of trade, the United

States required that its trading partners agree to on-site inspection by outside inspectors with "access to all places and data necessary" to ensure that the peaceful use guarantee was being observed. Furthermore, the United States required that the recipient allow the U.S. to designate the facilities in which produced fissionable material in excess of the recipient's peaceful needs was to be stored with a United States option to purchase this excess material. Finally, the agreements provided for a varying degree of U.S. involvement in the recipient's decision whether to reprocess U.S.-supplied special nuclear material. In some cases the United States sought and received a veto over such a decision, but in others the U.S. only sought to approve the method by and facility in which the reprocessing was to occur.

These government-to-government agreements continue today with modifications from time to time. This concept whereby the governments of supplier and purchasing nations establish their program for cooperation is used by countries dealing with nuclear trade for peaceful uses. The government-to-government agreements are the legal instrument that authorizes nuclear trade between the signatory countries and which specifies the conditions of cooperation.

In an important development in the United States, the first time that any agreements were challenged in the courts involved agreements between the U.S. and Sweden and the U.S. and Finland. The court decided it was a political question and that the courts should not become involved in the question. This decision may be appealed to higher courts.

INTERNATIONAL AGREEMENTS

In 1957 the Treaty of Rome entered into force. Providing for nuclear cooperation among the members of the European Community and the establishment of EURATOM, the Treaty facilitated regional cooperation and permitted free transfer of nuclear materials within member states. The Treaty, however, also imposed significant obligations upon countries party to it.

Thus, the elements of an international regime governing nuclear trade were already in place when the IAEA came into being. As the "Atoms for

Peace" program envisioned, the IAEA was given substantial safeguards functions. Not coincidentally, the provisions of Article XII of the Statute which amplify the nature of these safeguards closely parallel those in United States bilateral agreements, and supersede the U.S. safeguards when a safeguards agreement is signed with the Agency.

While the safeguards functions of the IAEA have drawn the most attention in the past several years, the IAEA also was originally conceived to assist in the "development and practical application of atomic energy for peaceful uses throughout the world." Indeed, the Agency was given broad latitude to "perform any peration or service useful in research on, or development or practical application of atomic energy for peaceful purposes," including acting as an intermediary for the transfer of nuclear material, equipment, facilities and services between members of the Agency.

The Technical Assistance Program of the IAEA has provided support to countries through the years and marks an effort of the Agency to assist emerging countries with their nuclear programs. There has been an increased interest at the Agency in safety related matters as evidenced by the recent establishment of the International Nuclear Safety Advisory Group (INSAG). This is a concept that I had recommended in order to encourage greater international cooperation on nuclear safety. The first meeting of the group was held in Vienna, in March, 1985, with the following stated objectives:

1. To provide a forum for the exchange of information on generic nuclear safety issues of international significance.
2. To identify important current nuclear safety issues and to draw conclusions on the basis of results of nuclear safety activities within the IAEA and other information.
3. To give advice on nuclear safety issues in which an exchange of information and/or additional efforts may be required.
4. To formulate, where possible, commonly shared safety concepts.

Within a few years after the "Atoms for Peace" speech, the organizational and procedural outlines which still govern international nuclear commerce were in existence. Nations had established a pattern that the incentives of sharing the peaceful benefits of nuclear energy were sufficient to accept a multilateral mechanism with legal authority to monitor their conduct. The concept that compliance with an international obligation should and could be verified by such means as sending inspectors into the territory of other foreign nations was then both bold and novel. Even more remarkable is the rapidity with which the legal regime was established.

In the mid-to-late 1960s, however, the continued impasse in the disarmament talks and the growing diffusion of nuclear technology all served to make nations recognize that further actions were necessary. In 1964 the countries of Africa joined together to ban proliferation on that continent in the Declaration on the Denuclearization of Africa. Three years later a treaty banning nuclear weapons in Latin America was signed by 21 states principally in that region. In this treaty the signatories agreed to apply IAEA inspection to ensure that the goals are met. Also in 1967, the Outer Space Treaty which prohibited orbital nuclear weapons came into being. Most recently, eight South Pacific nations signed a South Pacific Nuclear-Free Zone Treaty.

On July 1, 1968, the Non-Proliferation Treaty (NPT) was opened for signature. Under the Treaty, which entered into force on March 5, 1970, nuclear weapons states agree not to transfer, assist or encourage non-nuclear states to acquire or gain control over nuclear explosive devices. Non-nuclear weapon states agree not to seek or develop such devices and to submit to IAEA inspection on all peaceful nuclear activities within their territories, under their jurisdiction or carried out under their control anywhere. In return, non-nuclear weapon states receive assurances that all parties (A) have the "inalienable right" to develop and use nuclear energy for peaceful purposes, and (B) should make available the benefits of peaceful applications of nuclear technology to the other parties "on a non-discriminatory basis." With the conclusion of the Non-Proliferation Treaty, the basic institutional regime governing the international exchange of nuclear materials and equipment was in place.

Events up to and through 1970 vividly illustrated the extent to which nations were willing to forego certain national options so as to achieve order and predictability in their efforts to procure an adequate energy supply. The institutional structure thus far imposed was the embodiment of a number of political, strategic, technical and economic factors. Whatever the mix involved for an individual state, a great number of countries of differing political persuasions felt it was in their own national interests to join together in the creation of a body of rules governing nuclear commerce among nations. Under this framework the fruits of this technology have been made available to unprecedented numbers of nations. The result was the most President Eisenhower could realistically have expected. The atom was successfully harnessed internationally to generate electric power, to advance medical research and treatment, to develop new industrial processes and techniques, and to assist in the management of the world's food supplies.

COMMERCIAL AGREEMENTS

After the development of the necessary international treaties and bilateral agreements, subsequent commerical agreements are necessary that set forth the details, particularly those of a financial nature, which govern the actual transfer of materials, technology, and fuel.

Many of these commercial contracts are written to favor the supplier company. While the contracts usually are acceptable and an orderly supply of the contract items has normally occurred, problems continue to arise on matters that the purchaser did not anticipate. Some of the disputes have involved substantial sums of money and have been resolved only after a very difficult period of disagreement. It is helpful in avoiding such difficulties to have a clear set of national requirements and criteria against which bids for materials and technology will be sought. It is also useful to have people who are familiar with the practices and contracts of the supplier country to provide technical and legal advice concerning these contracts. This assistance will often avoid difficulties in the future that can be timely and costly.

The Export-Import Bank of the United States as well as similar banks in other countries provide long-term direct credit and financial guarantee programs to assist with nuclear exports from their countries. Banks have a number of financing options that will vary depending on the competitive situation and the financing requirements. The financial plan offered by the supplier country and its companies is a major part of any decision to purchase.

SUPPLIER EXPORT LICENSES

In 1974 most major suppliers agreed upon the so-called "Zangger List," meant to implement Article III of the NPT. The inclusion of an item on this list meant that its export would trigger IAEA safeguards designed to ensure that these items were not used for the development of nuclear explosives and also to provide assurances that none of these items was re-exported without similar safeguards.

The Zangger List consultations were, in a very special sense, a forerunner of the discussions which became known as the London Suppliers Conference. The initial concerns of the nuclear suppliers found their first formal expression in the final declaration of the NPT Review Conference held in Geneva in May, 1975. This declaration, adopted by consensus, urged that common export requirements relating to safeguards be strengthened.

By January 1976, participants in the London Suppliers Conference had reached agreement on a broad number of fronts and exchanged letters which moved the level and comprehensiveness of some areas of the international legal regime substantially beyond that contained in the NPT. In these letters the major suppliers agreed to the application of IAEA safeguards on exports of material, equipment and technology and replicated technology to preclude their use in nuclear explosive devices, including those for peaceful purposes. They also agreed to apply restraint in the transfer of sensitive technologies and accept special conditions governing the use or retransfer of sensitive material, equipment, and technologies. Consistent with this, they pledged to encourage multinational regional facilities for reprocessing and enrichment. Finally, the suppliers agreed to require physical security measures on exported nuclear facilities and materials.

Concurrent with these developments, individual states began taking actions designed to restructure international nuclear cooperation. After the Indian explosion in 1974 which used plutonium generated in a Canadian supplied research reactor, Canada undertook to renegotiate all of its existing agreements for cooperation to make clear its prohibition of peaceful nuclear explosives built or constructed with or through use of Canadian nuclear exports. The inability of Canada to conclude such strengthened agreements with India and Pakistan resulted in the suspension of all nuclear trade with those coutries.

Canada has not been alone in this approach. The United States also has re-examined the conditions under which it will supply nuclear equipment and materials. Prior to 1975, export licenses were issued by the Atomic Energy Commission on a routine basis. The issuance of a license was a ministerial rather than a policy action, its principal purposes being to register shipments and ensure compliance with the United States international obligations. During the past several years, however, the U.S. Congress has begun playing a more active role in the exercise of its oversight and legislative functions, and the Nuclear Regulatory Commission has started scrutinizing at the Commission level all controverisal and some not-so-controversial exports.

The Nuclear Non-Proliferation Act of 1978 imposed a uniform set of criteria on all nuclear exports from the United States without the benefit of broad international discussions. The criteria applied initially contain the guidelines accepted at the London Suppliers Conference and make them applicable to all material derived from U.S.-supplied material and equipment. Additionally, the United States now requires prior U.S. approval over the reprocessing and retransfer of U.S.-supplied and derived material. Along with other criteria concerning non-proliferation goals, reprocessing is only permitted where it occurs under conditions which give the United States "timely warning" of any diversion of the materials for explosive purposes. All U.S. non-nuclear weapon trading partners would be required to adopt full fuel cycle safeguards. While the President may suspend the application of one or all of these criteria to a particular export because of the individual circumstances involved, the process is complicated, cumbersome and replete with uncertainty.

Whatever its shortcomings, the existing framework of bilateral and multilateral control is a significant factor in international nuclear commerce.

DOMESTIC REGULATORY REGIME

Domestically, most nations provide for a legal regime to regulate nuclear power. The goal of these legal systems is the protection of health and safety and the environment. The regulatory systems also regulate, in conjunction with treaties and international agreements, the export and import of nuclear supplies and technology.

Laws and regulations promulgated for domestic legal systems govern licenses issued for nuclear power station construction and operation, transportation of nuclear materials, possession and use of by-product and special nuclear material, liability to the public resulting from injury to persons or to property due to nuclear accidents, and radiation exposure standards for public and workers in the nuclear industry.

In British Commonwealth nations, the nuclear power plant licensing process is centered on the inquiry system. A high court judge presides over an inquiry into the safety and environmental aspects of the proposed nuclear power plant. The inquiry usually lasts a year or more, during which time a record is compiled. The presiding judge then presents his final report and recommendations to the government. A recommendation is then made to the appropriate ministry, which has the authority to cancel plans for a nuclear power plant or to proceed with the project. An inspectorate monitors plant construction and operation if the projects are commenced. The Sizewell Inquiry in the United Kingdom exemplifies this process.

The regulatory scheme in the United States is administered by the U.S. Nuclear Regulatory Commission (NRC). Research on nuclear power plant safety issues and analysis of operating experience are performed by NRC. From these results, standards are written for matters such as quality assurance and general design criteria. Proposals for nuclear power plant construction and operation are then reviewed by NRC staff, using the standards as guidelines. Upon a favorable NRC review, a commitment is made to license the proposed facility for construction or operation. During plant construction and operation, NRC inspects the plant to ensure that its standards are being met. If the inspections find that standards are not complied with, enforcement action may be taken by NRC against the licensee.

Transportation of nuclear materials is regulated in the U.S. by NRC in cooperation with the U.S. Department of Transportation. Containers used for shipping must be constructed in accordance with specifications based upon International Atomic Energy Agency requirements to prevent release of radiation in a variety of accident conditions. Routes for road and rail shipment of nuclear material are prescribed. Security measures for safeguarding radioactive shipments are also included in the regulations.

Regulations provide requirements for possession and use of special nuclear material. Their purpose is to ensure that medical or industrial applications of radioactive material will be controlled by competent technicians who are responsibile for using the material safely. Accounting systems are established so the material can be traced to prevent its diversion for unauthorized purposes.

In most countries, a law is passed to provide an indemnity scheme to compensate members of the public whose person or property is injured as a result of a nuclear accident. The liability limit under this law in the United States increases with each new plant and as of this time is $635 million, which is available to pay parties injured in an extraordinary nuclear occurrence.

Radiation protection standards in most nations are based on the criteria developed by the International Committee on Radiation Protection (ICRP). ICRP has published standards to limit occupational and public radiation doses. These standards are universally recognized by nuclear power nations.

INSTITUTIONAL ARRANGEMENTS FOR INTERNATIONAL EXCHANGES

A system has developed that provides effective means for exchanges between countries in the scientific, technical and sometimes political areas. In the scientific and technical commmunity, nuclear societies have been established in most countries that permit cooperation with nuclear societies in other countries. The American Nuclear Society (ANS) has formal agreements with many nuclear societies in the world for the exchange of scientific literature and information. The nuclear societies have further organized under the umbrella of the International Nuclear Societies Group (INSG) as a forum to exchange views and information so as to avoid conflicts and build on the work that others are doing.

With public information, many countries have set up organizations, such as an industrial forum, to work on matters of public acceptance, educating members of the public, and dealing with members of the government. The United States has the Edison Electric Institute (EEI), the Atomic Industrial Forum (AIF), the American Nuclear Energy Council (ANEC), and the Committee for Energy Awareness (CEA) -- all of which cooperate with similar organizations in other countries.

The Institute for Nuclear Power Operations (INPO) is a mechanism to evaluate industry's performance in the construction and operation of nuclear power plants. It has an international advisory group so that cooperation with other countries on construction and operational issues can occur.

The Electric Power Research Institute (EPRI) also deals with specific projects and questions on an international basis. The industry has recently developed an organization to deal with human factors in nuclear power plants called the Nuclear Utilities Management and Human Resources Committee (NUMARC) and is considering expanding this to work on various technical hardware questions. It may be possible that international exchanges can be established with this organization as well.

A number of organizations undertake the writing and development of agreed standards on nuclear subjects. These include such organizations in the United States as the American Nuclear Society (ANS), the Institute of Electrical and Electronics Engineers (IEEE), the American Society of Mechanical Engineers (ASME), and the American Society of Testing and Materials (STM). Mechanisms have been established to permit the use of these standards on an international basis.

From all of these institutions a significant amount of international cooperation occurs that enhances the exchange of information, particularly in a scientific and technical way. From time to time there appears to be a need for some major exchanges, such as occurred in the early days of nuclear development at the Geneva Conferences. In the late 1970s the International Nuclear Fuel Cycle Evaluation (INFCE) resulted in extensive discussions internationally of major nuclear issues and technology.

On an ongoing basis, the many international nuclear meetings and nuclear publications permit a constant flow of knowledge. One of the difficulties is the use of the vast range of knowledge in a systematic and integrated fashion. The next significant international need could well be the employment of institutional mechanisms to permit an integrated approach to nuclear power questions.

CONCLUSION

The governmental, commercial, and technical institutional requirements and arrangements for international nuclear commerce are relatively complex but have through the years served as a good foundation for the transfer of materials and technology. In the process there have been disagreements and difficulties but on balance purchasing countries have been able to acquire the benefits of peaceful nuclear uses in an orderly manner. The result has been significant benefits to countries in terms of clean, efficient energy, as well as other peaceful uses such as medical and industry applications. From time to time uncertainties have emerged, but strong efforts to stabilize conditions have also occurred in order to give as much certainty to international nuclear commerce as possible. While certain issues remain, a strong effort is underway to maintain assurance of supply and stability in international nuclear commerce. International agreements, domestic regulations, and organized exchanges help assure that an orderly and safe introduction of nuclear power can be accomplished.

executive officers, etc. A wide variety of people go to workshops, sit down, and work on the actual problems they are having. This exchange is probably one of the most valuable of the Institute.

INPO helps us in our emergency planning. We have very strict emergency planning requirements, and they have formed an agreement with each company so that, if we had another incident like Three Mile Island, all the resources from one utility to another would be available through INPO in a very rapid manner.

In summary, INPO has become a vital partner in the nuclear industry's quest for better management. We work with them in my company every day. I have been head of an industry review group in the events analysis group for a number of years. I do participate in exchange visits and things like that. You cannot achieve high safety through regulation alone. You have to be striving to do more than that.

INPO has some teeth. By that I mean, we members realize that we're all our brothers' keepers, so if INPO should find a utility that is not trying to do well, that does not have programs to exceed minimum requirements, that is not making progress, INPO is authorized to withdraw their INPO membership. If a single utility ever had its membership withdrawn, one can imagine the consequences from the board of directors of that company, from the nuclear insurers, from the government, from the press. Nobody could take that type of heat. So that's just about as strong an arm-twister as you can imagine.

I'll be looking forward to discussion in more detail, the INPO situation, with you. I'm very proud of it. I'm proud of my association with INPO. I think it is one of the most forward-looking organizations we have in the nation.

which we call nuclear network. It is an electronic mail system, where all the members in various categories are interconnected. About 30 messages a day are passed back and forth. If a person has a problem with a piece of equipment, he might get on the typewriter and say, "Have you had this problem with this type of valve?" Some guy will say, "Yeah, I had it last week and it was the whigamajig." And so that helps everybody solve the problems as they come along. In conjunction with this type of problems, we have a nuclear plant reliability data system. We designated certain items of equipment for which detailed engineering data must be provided. We fill out cards, computerized cards; then we put each one of those in a data bank. For our plant we have something like 9,000 entries, 9,000 different pieces of equipment in a data bank. All the other utilities do the same. Then, whenever something happens to that piece of equipment, we'll fill out a failure card, a problem card. When we encounter a new problem, we access the data bank to see what happened in someone else's plant. Or, if we want to get a more reliable component, we find out which one's not having problems. That's called the nuclear plant reliability data system.

The fourth major area in which INPO is involved is a simple one of technical assistance and technical exchanges. This is emphasized to international participants as well. INPO performs special assistance visits in almost any area in which they're involved. If somebody is having trouble with training, call up INPO and say, "Hey,I'm having trouble with requalification of operators. Can you send me an assistance team?" Shortly they'll send a couple of people down who have gone to many plants, and who are very able to diagnose the problem and help you find out how to tackle the problem you have. From their various evaluations and assistance visits, they gather information and create documents called "Good Practices". A way, not necessarily the only way, but a way to do a certain job in a plant. So if you're having problems on the job you can pull out a "good practice" and see if you want to use that.

Workshops. There are workshops for chemists, radiation protection people, maintenance superintendents, operations superintendents, chief

NUCLEAR NETWORK TOPIC LISTING

ID	Topic Title
BW	B&W Owners Group
CE	CE Owners Group
CI	Coordination With INPO
DC	Design, Construction & Preoperational Testing Information Exchange
EM	Exchange of Miscellaneous Information
EP	Emergency Planner Information Exchange
FS	Fire Protection & Plant Security
GE	BWR Owners Group
HL	Emergency Hotline
IC	International Coordination Exchange
IA	INPO Significant Event Reports - 1980
IB	INPO Significant Event Reports - 1981
ID	INPO Significant Event Reports - 1982
IS	INPO Significant Event Reports - 1983 to date
MA	Meeting Announcements & Summaries
NP	NPRDS Information Exchange
NR	Nuclear Records Management
NT	NUCLEAR NETWORK Training
OE	Operating Plant Experiences - February 1981 to date
OM	Operations & Maintenance Information Exchange
OR	INPO Operations & Maintenance Reminders
PS	NRC Daily Plant Status Report
QA	Nuclear Quality Assurance Information Exchange
RI	Regulatory Information Transmittal
RP	Radiological Protection & Chemistry
TS	Training & Staffing
WO	Westinghouse Owners Group
GP	Good Practices from INPO Evaluation Reports
SR	Operational Reactor Safety Review Information Exchange

NUCLEAR NETWORK

- FLEXIBLE, TIMELY COMMUNICATIONS SYSTEM AVAILABLE TO NUCLEAR UTILITIES WORLDWIDE

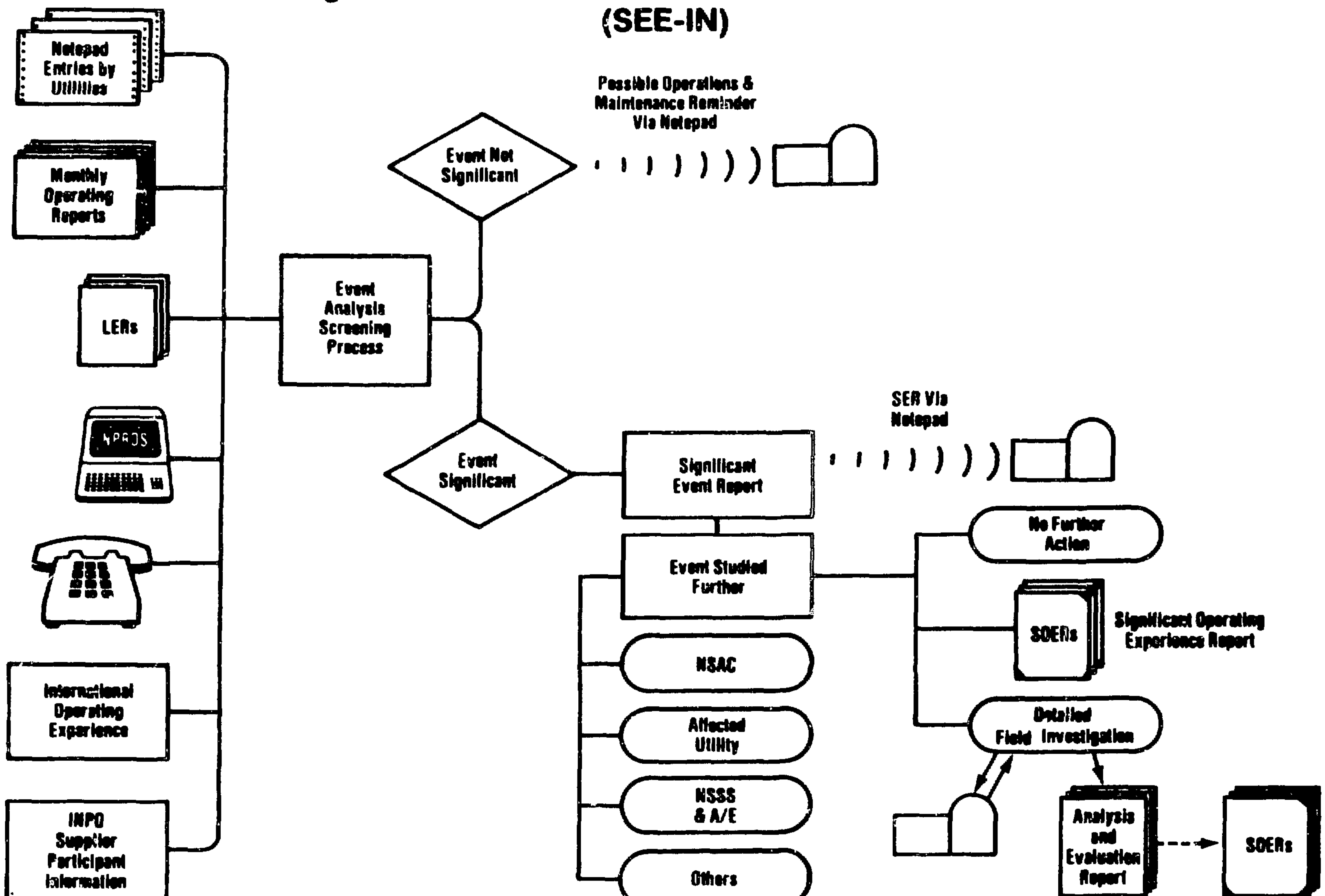
- 95 ORGANIZATIONS BELONG TO NUCLEAR NETWORK, REPRESENTING 14 COUNTRIES. 500 INDIVIDUAL PARTICIPANTS HAVE ACCESS TO THE SYSTEM

- 1400 MESSAGES PER MONTH ARE PLACED IN NETWORK

- 25 TOPIC AREAS CAN BE SELECTED

- MESSAGES CAN BE EITHER PUBLIC OR PRIVATE

Significant Event Evaluation and Information Network (SEE-IN)



THE ROLE OF THE INSTITUTE OF NUCLEAR POWER OPERATIONS

Robert P. McDonald
Senior Vice-President
Alabama Power Company

I would like to provide you with an overview of how our industry has collectively joined together to enhance the management of nuclear power plants through their enterprise of the Institute of Nuclear Power Operations (INPO). I would then follow with a discussion of utility management principles as they apply in each utility.

As you know, there were many lessons learned from Three Mile Island. To us in the nuclear power industry, two stood out. In the two that stood out to us -- now I say us because in some of the other lectures you find other things that stand out to other people -- but to us in the nuclear power management industry, two things stood out. First was that mere compliance with regulatory requirements cannot adequately manage a nuclear power plant. There has to be something in addition to meeting minimum requirements. Second, an accident at a single nuclear plant any place in the world has a rippling adverse effect upon all the other plants in the world. Now I might say that those were the two adverse effects. I do believe that the event itself has prompted a restudy, a rethinking, a revitalization of the intensity of management; and in the long run it will probably turn out to have been of benefit to all of us. In any case, these two lessons prompted us to form the Institute of Nuclear Power Operations as a focal point and facilitator for all utilities to strive for higher levels of performance. Now I say as a focal point and a facilitator because INPO does not have management authority. They are supported entirely by membership fees from the members. The Board of Directors is made up entirely of officers from companies that are members. So they are in essence a management service company for us. But we are very careful to make them very, very independent, answering to no single utility. All 55 utilities in the United States having nuclear power plants are members of INPO. INPO also has 13 members from other places in the world, and I see a few of you here today who are not presently members, and I hope to meet you some time as members of INPO at some of the INPO functions.

"THERE MUST BE A SYSTEMATIC GATHERING, REVIEW AND ANALYSIS OF OPERATING EXPERIENCE AT ALL NUCLEAR POWER PLANTS, COUPLED WITH AN INDUSTRYWIDE INTERNATIONAL COMMUNICATIONS NETWORK TO FACILITATE THE SPEEDY FLOW OF THIS INFORMATION TO AFFECTED PARTIES. IF SUCH EXPERIENCES INDICATE THE NEED FOR MODIFICATIONS IN DESIGN OR OPERATION, SUCH CHANGES SHOULD BE IMPLEMENTED ACCORDING TO REALISTIC DEADLINES."

KEMENY COMMISSION RECOMMENDATION

OPERATING EXPERIENCE

THE NEED FOR AN IMPROVED MEANS OF ANALYZING AND DISSEMINATING OPERATING EXPERIENCE WAS ONE OF THE MAJOR LESSONS OF THE THREE MILE ISLAND ACCIDENT.

THREE ORGANIZATIONS WERE CHARTERED TO ACCOMPLISH THIS TASK:

- NUCLEAR SAFETY ANALYSIS CENTER (NSAC)
- INSTITUTE OF NUCLEAR POWER OPERATIONS (INPO)
- NRC'S OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA (AEOD)

In addition to utilities and people who operate plants, INPO has 13 members who are reactor-vendors -- Westinghouse and General Electric, etc. -- or who are architect engineers, like Bechtel, Sargent and Lundy and that type.

Stated a little different way, INPO was formed to assist us to get high levels of safety and reliability in operations. In forming INPO, the chief executive officer of each and every company committed himself to fully support what was to be done through INPO. The organization is headquartered in Atlanta. It has about 400 people. Its staff is composed primarily of well-seasoned professional people. About 1/4 of its technical staff are loanees from members. For example, my company, Alabama Power, has had a member there, one or two members, for the past three or four years. It's usually someone whom I take out of a job in the plant. The last two or three have been shift supervisors, the senior job on shift. We've taken him out of the plant and sent him to INPO for a year to a year and a half. Well experienced people, professionals.

INPO has four major areas in which they work, and I'm giving you this overview, but I've also brought some books with me. Like this. It's called <u>The Institutional Plan for the Institute of Nuclear Power Operations</u>. It describes INPO in some detail. In this you will find that INPO has four major areas of work. The first is the conduct and evaluations of each plant and each corporate organization involved. The second is assistance with training, and accreditation of training. The third is analyses of events, and coordinated with that exchange of information. Fourth, assistance with exchanges, that is, visits.

Let me describe each one of these briefly. I hope by my describing them to prompt some questions from you about what you might be interested in in more detail, because now I am just giving you an overview.

In the first area, the evaluations -- evaluations, inspections, audits, reviews, analyses -- call them whatever you want -- they are carried out in light of published performance, objectives, and criteria in eight different areas. Now you might say, "Oh, that looks like a standard, a regulation." But it's not. These are written to assure that

U. S. NUCLEAR INDUSTRY'S OPERATING EXPERIENCE PROGRAM:

SIGNIFICANT EVENT EVALUATION AND INFORMATION NETWORK (SEE-IN)

CREATED IN 1979 BY NSAC; EXPANDED TO A JOINT NSAC/INPO PROGRAM IN 1980; SHIFTED TO INPO IN 1982.

MAJOR ELEMENTS:

- EVENTS ANALYSIS
- NUCLEAR PLANT RELIABILITY DATA SYSTEM (NPRDS)
- "NUCLEAR NETWORK" SYSTEM
- PERFORMANCE DATA COLLECTION AND TRENDING
- LICENSEE EVENT REPORT AND NRC DATA BASES

you have good programs for achieving more than minimum compliance and that you are active and being successful in the pursuit of those programs. They have no minimums in them. The evaluation is done against these. Each evaluation takes about two weeks. The evaluation team, again, is of specially trained experts, from 10 to 20 people involved in one evaluation. INPO is now starting its fourth cycle of plant evaluations. After each evaluation, INPO provides a written report giving its findings. In its findings it makes recommendations for utility action to overcome particular problems. The utility looks at them, meets with INPO, and arrives at an agreement on what they think they should do. That's printed in the report. Then the report is formally sent out. It is not sent out nationwide or industry-wide. It is sent to that utility only, and that utility proceeds to implement those improvements. INPO then follows up in its next evaluation. When they come in for the next evaluation, that's the first thing they look at. "Did you act on what we found last time and what you said you were going to do?" Then sometimes they say, "Did you do them right?"

The evaluations don't stop there. They also conduct evaluations of the corporate offices of each utility. They have another team, a similar activity, with a similar set of objectives. This has to do with <u>me</u>. They come and evaluate me and the people who work in the general office, to see what I am doing to support and manage those facilities. The same type of report is generated. In each of these evaluations, the team consists of a few people from other plants. That keeps them down to earth. It provides a way to exchange experiences, and it provides good practices by word of mouth from one to another.

The second area involves training. I think all of us have realized from our first days as managers that training is of vital importance. INPO assists people in training by developing training programs and assessing the needs of personnel. They also develop guidelines which the industry can use, and they administer an industry-wide accreditation program. Now, accreditation program means this -- from the industry guides developed by INPO, we each develop training programs based upon a systems approach which is performance-based. By that I mean that you look at a job, you see what is exactly required at each task in that job. Then you put all those tasks together into position, like a reactor operator, a chemist, a health physicist, you put them into a job. Then you design a course around those task and job requirements, put it in a curriculum, teach it in a classroom with knowledgeable people, give examinations, pass, and so forth. When they get out, then they go back to the job. You look and see how those people do on the job. Do they make mistakes? Are there some areas where they were insufficiently trained? You feed that back into the program, called performance-based. Each of the courses taught by individual utilities is then accredited by INPO. They have an accreditation team that comes and reviews the programs and results and accredits those training programs just as you would accredit a university or college. A very prestigious-type board. So we're going about training in a very systematic way. To give you an idea of how systematic: 1/6 of all our skilled people are in training at any one time. If you have 1/6 of your people in training, that means that you must have people to train that 1/6th, and we in one plant have a training staff of about 32 people. It's a big building, and we usually have as many as eight or ten classes going on at one time. It is a very large effort.

The third area of responsibility for INPO is event analysis. INPO analyzes the events that occur in operating nuclear plants, in an effort to search out those things that are precursors or symptoms of a significant problem that could happen next time. Last year they analyzed some 9,000 such events, and out of those 9,000 events, they found 15 that were truly significant. Those 15 were analyzed in depth, and then a report sent to all the members talking about the results of that analysis and making recommendations on what each company generically could do to decrease the probability of that leading to a serious event in its plants. When INPO sends those out, the individual utility takes them, works with them, implements the recommendations. INPO checks on the utilities' implementations through the evaluation process.

In conjunction with this type of transfer of information, INPO manages a teleconferencing system, computer teleconferencing system,

XII THE ROLE OF THE INSTITUTE OF NUCLEAR POWER OPERATION (INPO)

Robert P. McDonald
Senior Vice President
Alabama Power Company

XIII NUCLEAR SAFETY ACTIVITIES OF THE INTERNATIONAL ATOMIC ENERGY AGENCY (IAEA)

Morris Rosen
Director, Division of Nuclear Safety
International Atomic Energy Agency

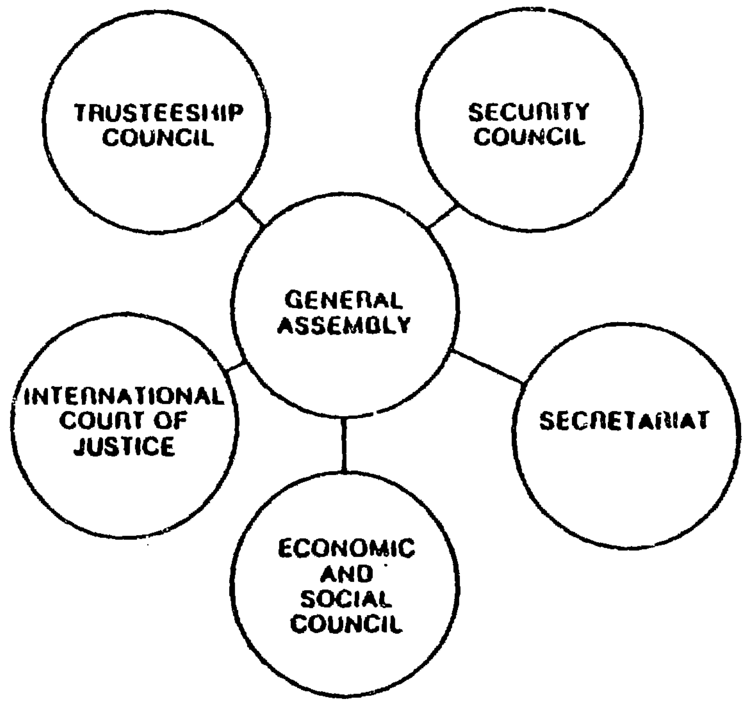
NUCLEAR SAFETY ACTIVITIES OF THE IAEA

Dr. Morris Rosen
Director, Division of Nuclear Safety
International Atomic Energy Agency

Let me say welcome to many of the old friends I know here and some of the new friends that I hope I will have in the future. I think some of you may have heard some of my talk before, but there are new programs, and I think these new programs that I will introduce today you will hear for the first time. I'll talk about the agency -- what it is, what it does, how it works. I'll also talk a bit about the nuclear power situation in the world, just to give you a little perspective.

Let me start by telling you what the agency is, or perhaps I should start by saying what it is not. The agency is not part of the UN. It is part of the UN system, and if you look at this first chart, it shows the UN. And the UN, as most of you are aware, is a General Assembly composed of over 160 countries. There are five major organs to the UN, the Secretariat and the Security Council being the ones that you are probably most familiar with. Attached to the UN are a number of more minor organs, and these organs or bodies usually begin with the words United Nations. If you look at the chart, you'll see the UNDP, United Nations Development Program, United Nations Environmental Program -- these organizations are part of the UN. They have the same membership; their budget comes from the UN. They do not have an independent budget system; it comes from the UN. And you can usually tell these other United Nations organs by the use of the words, "United Nations". They are normally the first two words in the title of the organization. There are a number of other bodies which are part of the UN system, but are not part of the United Nations, and the next chart shows these organizations. When I say they are not part of the UN, I mean they have their own membership, which is different from the UN. For example, the agency has 111 members, and the UN has over 160. They have their own budget, so they are not dependent on the UN for their budget. Their member states contribute the budget. These organizations normally begin with the words like "International" or "World", and you can see it in some of the

The United Nations System



UNITED NATIONS ORGANS

UNDP	UNITED NATIONS DEVELOPMENT PROGRAMME
UNEP	UNITED NATIONS ENVIRONMENTAL PROGRAMME
UNDRO	UNITED NATIONS DISASTER RELIEF ORGANIZATION
UNU	UNITED NATIONS UNIVERSITY
UNITAR	UNITED NATIONS INSTITUTE FOR TRAINING AND RESEARCH
UNCTAD	UNITED NATIONS CONFERENCE ON TRADE AND DEVELOPMENT

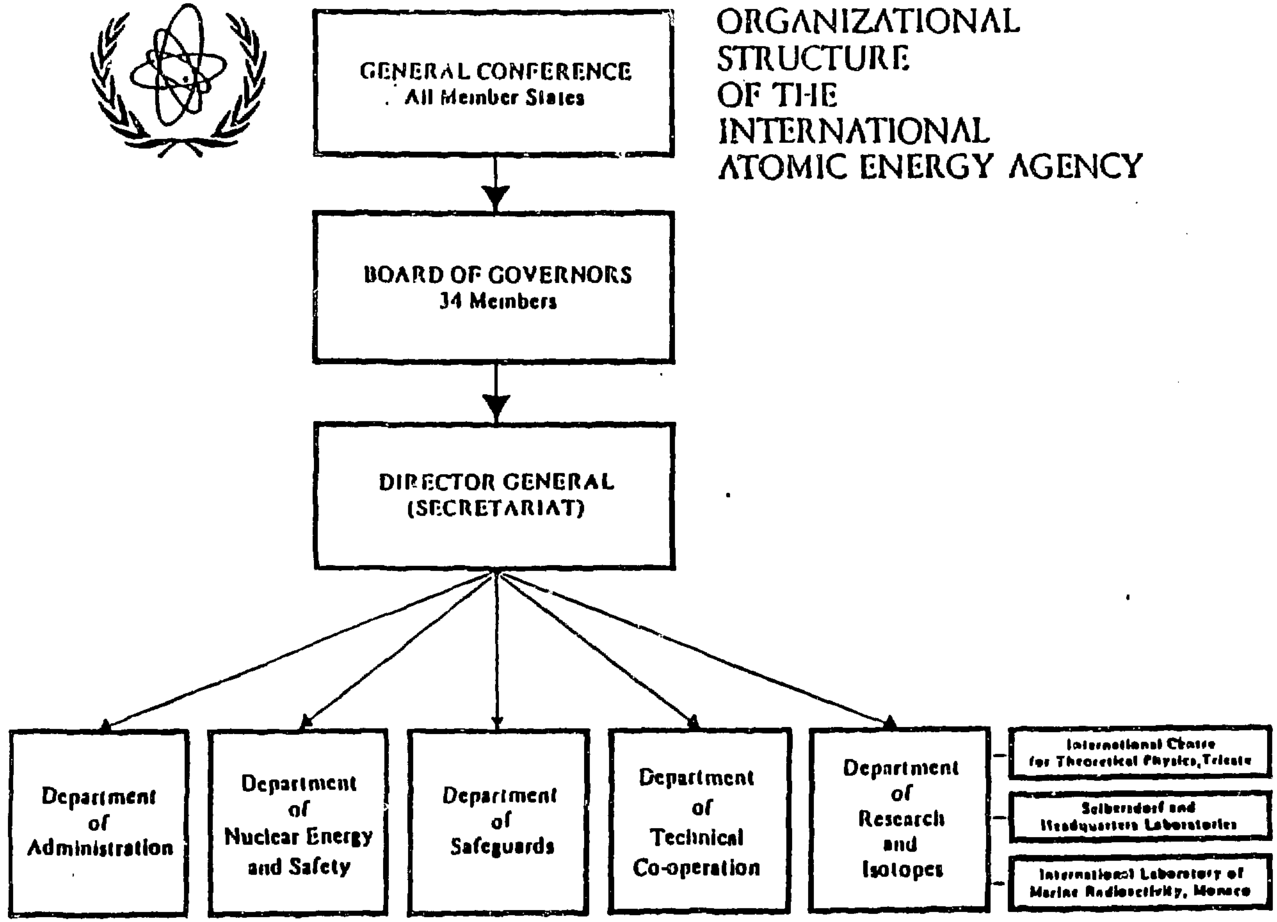
SPECIALIZED AND AUTONOMOUS AGENCIES

IAEA	INTERNATIONAL ATOMIC ENERGY AGENCY
ICAO	INTERNATIONAL CIVIL AVIATION ORGANIZATION
ILO	INTERNATIONAL LABOR ORGANIZATION
IMF	INTERNATIONAL MONETARY FUND
IMO	INTERGOVERNMENTAL MARITIME ORGANIZATION
WHO	WORLD HEALTH ORGANIZATION
WMO	WORLD METEOROLOGICAL ORGANIZATION
UNESCO	UNITED NATIONS EDUCATION, SCIENTIFIC AND CULTURAL ORGANIZATION
UNIDO	UNITED NATIONS INDUSTRIAL DEVELOPMENT ORGANIZATION

agencies I have listed, the International Atomic Energy Agency, the International Civil Aviation Organization. If you look toward the end, there are two there that do begin with the words "United Nations", and that's because originally these were part of the UN and then became independent, autonomous bodies. The latest addition to the independent bodies is the last one on the list, which is UNIDO, the United States Industrial Development Organization, which is just becoming independent I believe this month, or has within this month. By becoming independent, that organization now has its own membership, will elect its own Director General, and will have its own budget. So there is a clear distinction between these autonomous and specialized agencies and the United Nations itself.

Let me show a little closer picture of the IAEA. If we start on this chart, you see the General Conference, and that is the membership of the IAEA, 111 countries, and the last country was China, which was just admitted at the beginning of the year. China became the 111th member of the IAEA. Now running the IAEA is a Board of Governors. It's now composed of 35 members; China is now on the Board of Governors. The Board of Governors acts like the executive body or the board of directors of the organization. It really sets the program, sets the budget and how to spend the budget. The Board of Governors meets about three times a year, and there is one governor from each of the countries on the board, and the governor normally has a staff. Some of the staff are located in Vienna in missions to the IAEA, so many countries have missions located in Vienna that support the governor in running the IAEA. Reporting basically to the Board of Governors is a Director General, and Dr. Hans Blix is current Director General. He has been Director General for almost four years. He has just been nominated by the Board of Governors to be the Director General for another four years beginning December of this year. That recommendation of the Board of Governors must be approved by the General Conference, and the General Conference will meet in September, and it is more than likely that the General Conference will appoint Dr. Blix Director General for another four years.

ORGANIZATIONAL STRUCTURE OF THE INTERNATIONAL ATOMIC ENERGY AGENCY



The Director General is the chief of a staff of about 1600 employees, approximately half of which are professional staff. The employees come from about 60 countries of the 111 member states. The agency itself is broken into five major departments. The Department of Administration does, as in most organizations, the administrative work, but it also in this case has the translation, interpretation, languages in that particular area. The next department is the one that many of you will be concerned with -- the Department of Nuclear Energy and Safety and the Division of Nuclear Safety is in that department. The Department of Safeguards, which is the third one, is the biggest in the agency, and its basic responsibility is associated with the NPT, the Non-Proliferation Treaty. It has almost 400 professional staff, and it was the fastest-growing organization, department, at the agency within the past several years. Another department which you will also have much to do with is the Department of Technical Cooperation. It is the department which administers the technical cooperation fund of the agency, which are funds earmarked basically for the developing countries. The budget of the agency, the regular budget, is approximately $100,000,000. In addition to that regular budget there is a budget basically assigned to this department. It is a voluntary budget. It is voluntary funds. It amounts to over $30,000,000, and that voluntary fund is distributed to the developing countries on a variety of projects, and I will show you in the safety area how we utilize the funds in the Department of Technical Cooperation. On the political side, I should mention that the reason the Technical Cooperation funds are voluntary is the desire of the big contributors to basically have some control on how the money is spent. If it were part of the regular budget, then all the members could determine through the Board of Governors how the money is spent, and there would be some difficulty arising with the use of certain funds for the Non-Proliferation Treaty and the safeguards efforts. The last department on that slide is the Department of Research in Isotopes. It does much of the work in the agricultural area, the medical area, and it's another department with which the developing countries have some interaction.

DIVISION OF NUCLEAR SAFETY

RADIATION PROTECTION

- Basic Criteria on Radiation Protection
- Occupational Radiation Protection
- Radiation Protection of the General Public and the Environment
- Transport Radiation Safety
- Planning and Preparedness for Radiation Emergencies
- Handling of Radiation-Exposed Persons
- Physical Protection of Nuclear Facilities and Materials

SAFETY OF NUCLEAR INSTALLATIONS

- Safety Principles and Regulatory Activities
- Siting of Nuclear Installations
- Safe Design and Construction of Nuclear Installations
- Operational Safety of Nuclear Installations
- Safety Aspects of Quality Assurance
- Safety Research and Development

RISK ASSESSMENT

- Risk Analysis Techniques
- Comparative Risk Assessment
- Risk Perception

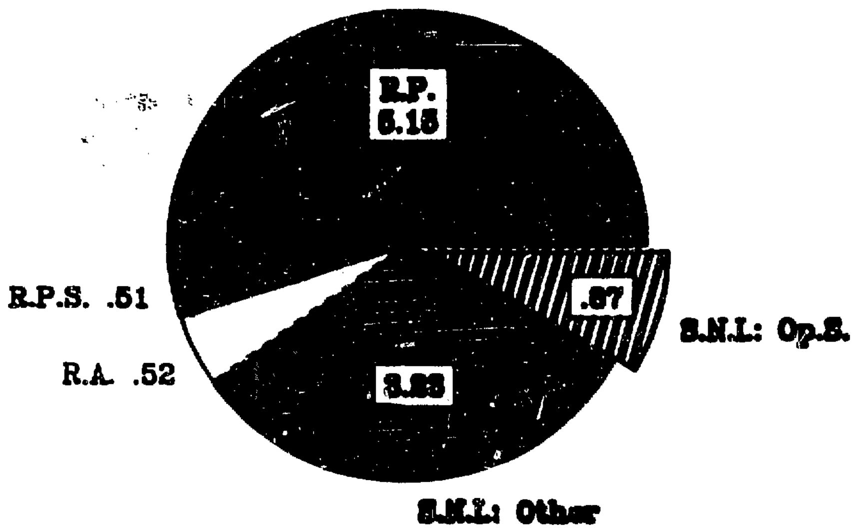
RADIATION PROTECTION SERVICE

- Headquarters & Seibersdorf
- Outside Assistance

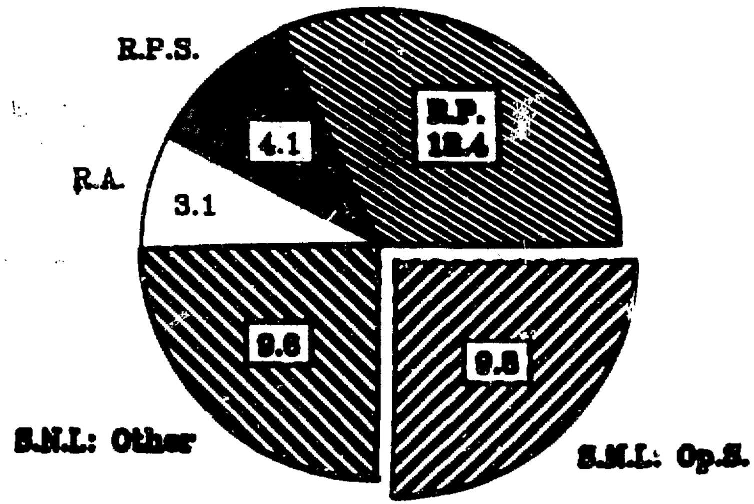
Let me show you a little more about the Division of Nuclear Safety. The division has basically four sections. It has one on radiation protection; another one on safety of nuclear installations, which has basically to do with the power plants and research reactors; another on risk assessment; and a fourth on radiation protection services. There are 18 sub-programs run by these sections. They're rather typical ones that you would associate with radiation protection and safety of nuclear installations. In the risk assessment area, we have a growing effort in risk analysis techniques. We have work on comparative risk assessment. By comparative, I mean comparative of various energy sources, comparison of coal, oil, nuclear. We are shifting that more towards the new area which is commonly referred to as risk management. We also have some work on risk perception, that's public opinion, but we are cutting that down also. It's a somewhat political area. It's relevant to a few countries which have significant problems in the public opinion area, but on the advice of the Board of Governors we are cutting that out, again, because it is a sensitive area, really peculiar to a few countries, the U.S. of course being one. In the radiation protection service area, we originally did radiation-protection services for agency staff, typically the safeguards inspectors and those who read the dosimeters and gave advice on radiation protection. We are now expanding that to outside services, particularly in Africa, where we are assisting many of the African countries and setting up radiation protection services for their own countries. As an initial step in many of these countries, we actually give them dosimeters which we take back and read at the agency until they can set up their own program.

If we take the overall technical assistance budget and the regular budget, the Division has available about $10,000,000/year. We have a professional staff of about 40. This particular slide shows the breakdown of the funding for the various operations. RP means radiation protection, SNI means safety of nuclear installations, RPS, the radiation protection services, and RA is risk assessment. Out of the 39 professional individuals in the Division, we now have 10 individuals dedicated only to operational safety. You'll see later some of the programs in operational safety where these individuals are now attached.

NUCLEAR SAFETY DIVISION
1986 RESOURCES INCLUDING TC

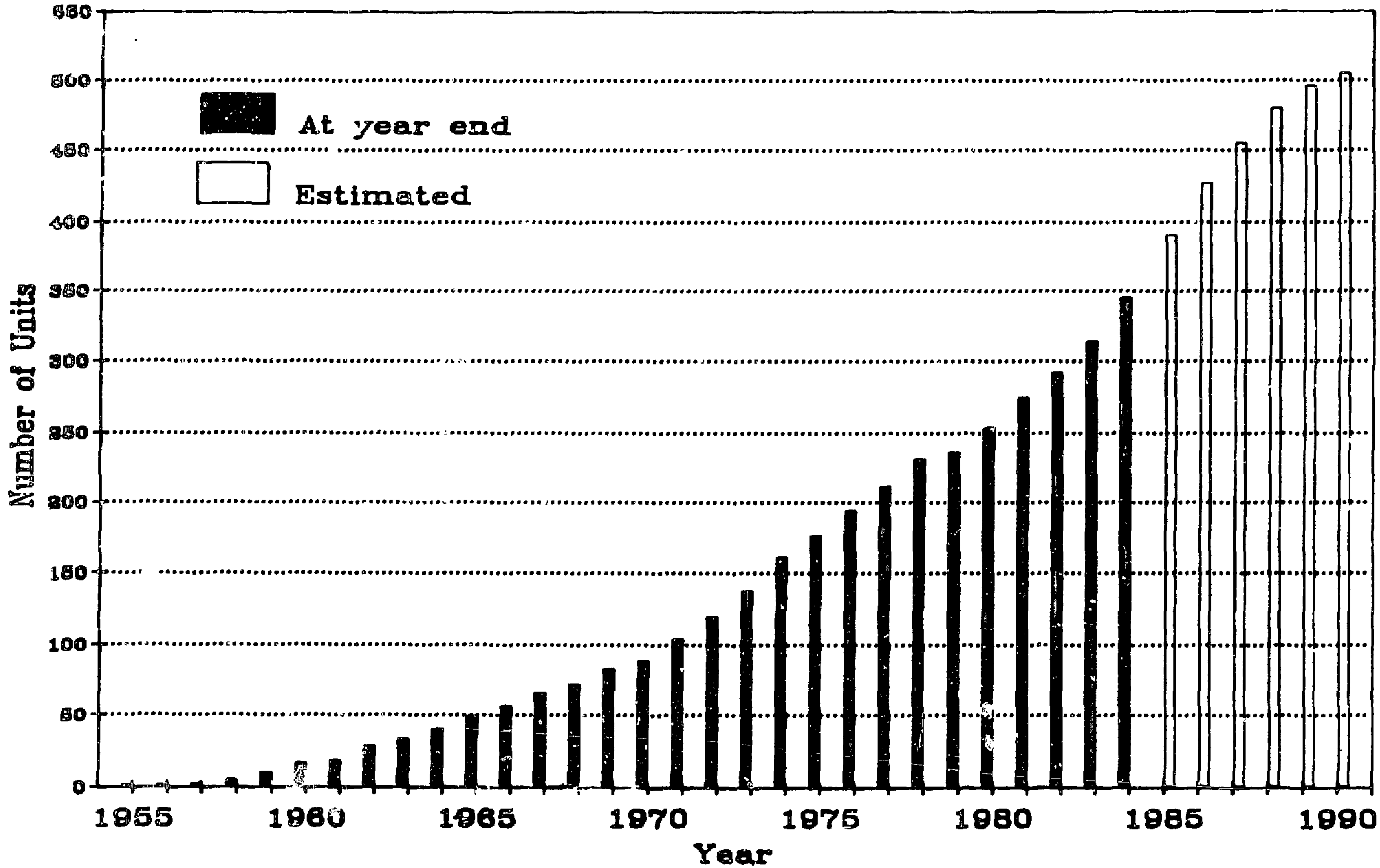


EXPENDITURE: $10.28 Million



PROFESSIONAL PERSONNEL (39)

Increase in the Number of Nuclear Power Reactors

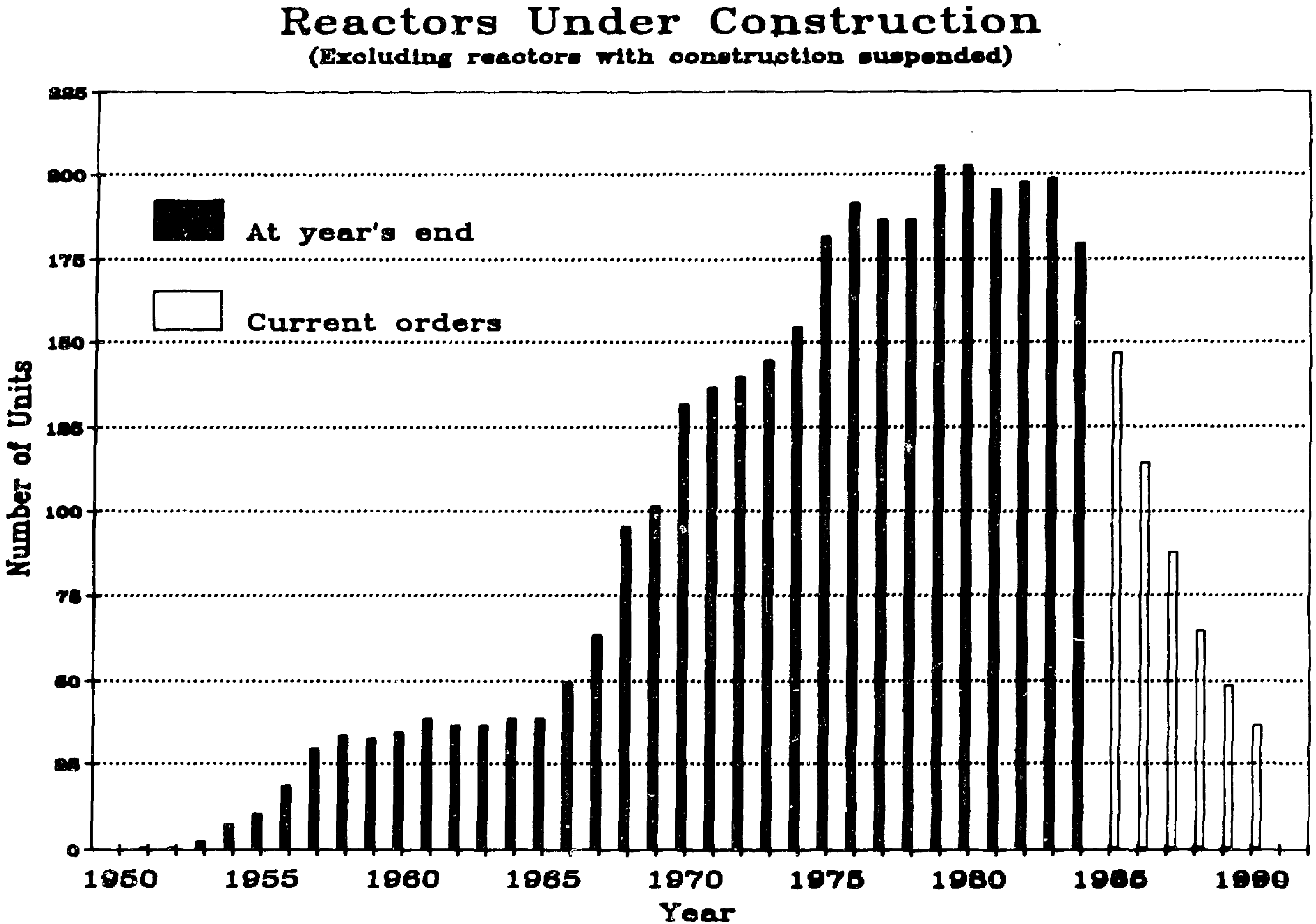


Source: IAEA Power Reactor Information System (PRIS)

Let me talk just a little about the world-wide situation in nuclear power. I don't know how much of this you've had over the last week or two, but normally I find it useful to give you some perspective as to what the nuclear power situation looks like. I'll start with the first chart, which just shows the increase in the number of nuclear reactors since the first one in 1954. You'll see that, by the end of 1984, there were 345 operating reactors. That now will grow at a fairly rapid pace in the next one or two or three years, with about 40 coming on line each year. We'll start tapering off around 1990, when we may have only 10 additional plants. In 1984 there were 34 new power plants put on line, six in the U.S., six in France, four in the FRG, another four in the Soviet Union. These 34, however, were only a net increase of 30, because there were four reactors decommissioned in 1984, one in Canada, one in France, one in FRG, and the Dresden reactor in the U.S.

While the number of plants still increases, the numbers under construction will show a dramatic decrease. At the end of 1984, there were 180 plants still under construction, and that's a decrease of about 30 from the peak, which occurred around 1980. These numbers, of course, will decrease quite dramatically in the coming years unless the orders for nuclear power plants take a turn for the better. And we will come down below 50 under construction towards the end of the '80s.

One more chart lists the countries with nuclear power plants in operation and nuclear power plants under construction. The total number of countries in operation or under construction is 32. There are 26 of these countries with plants in operation at the present time. South Africa was the last country to introduce an operating nuclear power plant. There are six countries which still have their first plant under construction: China, Cuba, Mexico, Philippines, Poland, and Romania. The list of 32 which are on this chart does not include Austria, which has the unique distinction of having a completed plant, completed since about 1978, and still, for political reasons, the plant is not in operation. The decision will probably be made within the next six months. This has been going on for many years, but it is getting to the point of no return. If the plant is delayed any further, it will probably take



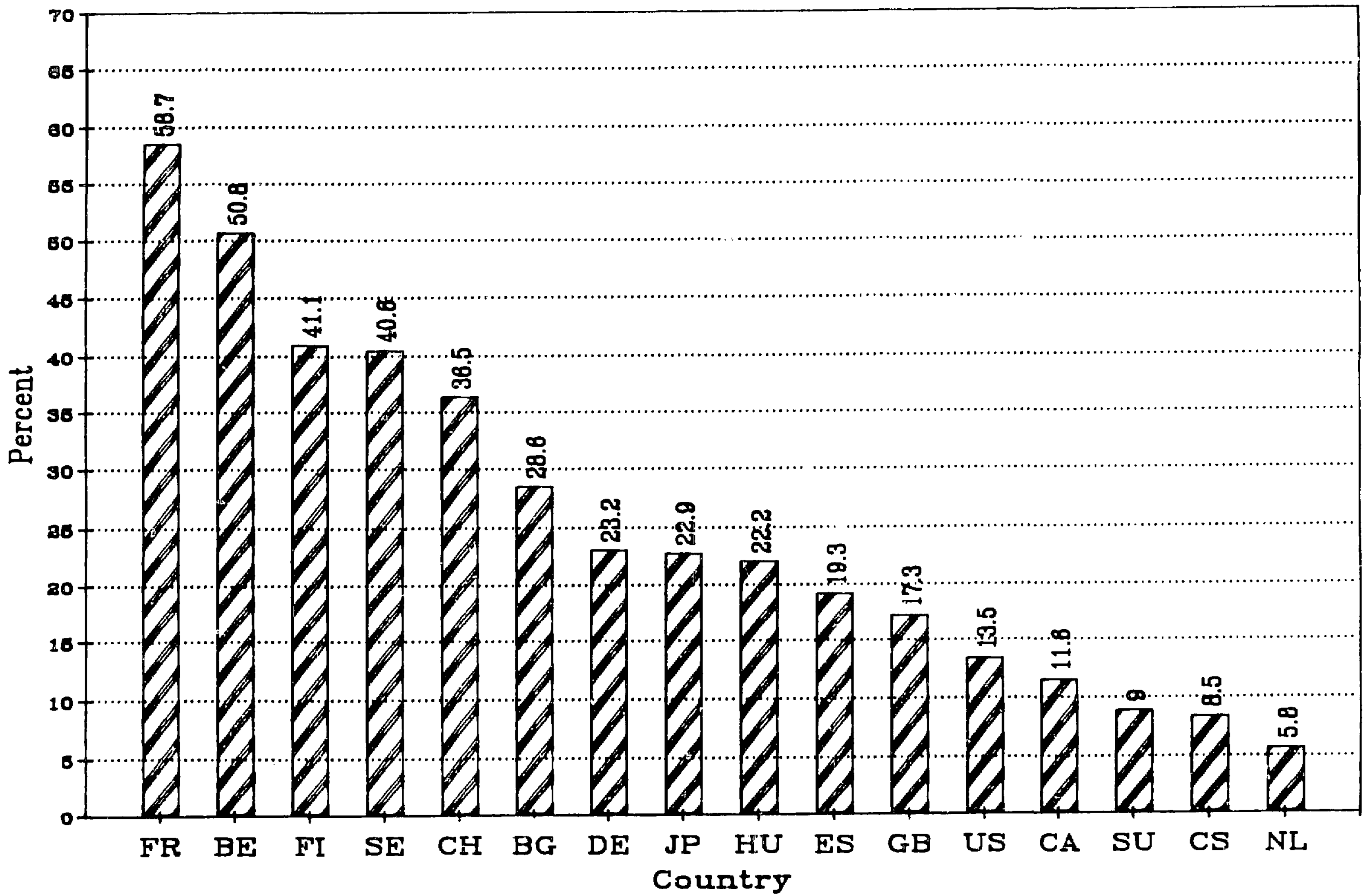
Source: IAEA Power Reactor Information System (PRIS)

NUCLEAR POWER PLANTS IN OPERATION AND UNDER CONSTRUCTION

COUNTRY	IN OPERATION No. of Units	Total MW(e)	UNDER CONSTRUCTION No. of Units	Total MW(e)
ARGENTINA	2	935	1	692
BELGIUM	6	3473	2	2012
BRAZIL	1	626	1	1245
BULGARIA	4	1632	2	1906
CANADA	16	9521	7	5630
CHINA	--	--	1	300
CUBA	--	--	1	408
CZECHOSLOVAKIA	3	1194	10	4394
FINLAND	4	2310	--	--
FRANCE	41	32993	23	28355
GERMAN Dem Rep	5	1694	6	3432
GERMANY, Fed Rep of	19	16114	7	6881
HUNGARY	2	805	2	820
INDIA	5	1020	5	1100
ITALY	3	1286	3	1999
JAPAN	31	21751	10	9182
KOREA, Rep	3	1790	6	5622
MEXICO	--	--	2	1308
NETHERLANDS	2	508	--	--
PAKISTAN	1	125	--	--
PHILIPPINES	--	--	1	620
POLAND	--	--	2	880
ROMANIA	--	--	3	1980
S. AFRICA	1	921	1	921
SPAIN	7	4690	3	2807
SWEDEN	10	7355	2	2100
SWITZERLAND	5	2882	--	--
TAIWAN	5	4011	1	907
U.K.	37	9564	5	3130
U.S.A.	85	68867	34	38242
U.S.S.R.	46	22997	39	36575
YUGOSLAVIA	1	632	--	--
TOTAL	345	219696	180	163448

December 1984

Percentage of Nuclear Power of Total Electricity Generated



two to four years to get the plant in operation. If it takes any longer, the plant will require an extensive amount of back-fitting which will probably make it impractical to start the plant.

Another figure showing the world-wide use of nuclear power is a figure which shows the nuclear power in several countries as a percentage of the total electricity generated. There is now about 13% of the world's electricity produced by nuclear power. There are a number of countries that lead in that, France being No. 1, with over 50% of its electricity coming from nuclear power. It's followed by Belgium, Finland, Sweden, Czechoslovakia, and Bulgaria. The Soviet Union is on our chart here. It has about 9% of its electricity produced from nuclear power.

A word about the age of nuclear power plants. There are many who still believe that nuclear power is a new industry, yet if you look at the figures, you'll find that there have been nuclear power plants operating for about 30 years, and quite a number that have operated over 15 years. In the chart, the next chart, it's been broken into five-year categories as to the countries and the number of plants operating in these five-year periods. If we look at the last one, which is more than 15 years, you'll find over 60 reactors that have been operating more than 15 years; another 80 that have been operating between 10 and 15 years; almost 80 between 5 and 10 years; and then almost 120 that have operated less than 5 years.

The next one shows the same information but perhaps a little more graphically. It does it with a bar chart showing the number of reactors that have operated a given number of years. I just use this to point out that nuclear power is not a new industry. If you look at the average lifetime now of power plants in operation, they have operated on average for over 10 years.

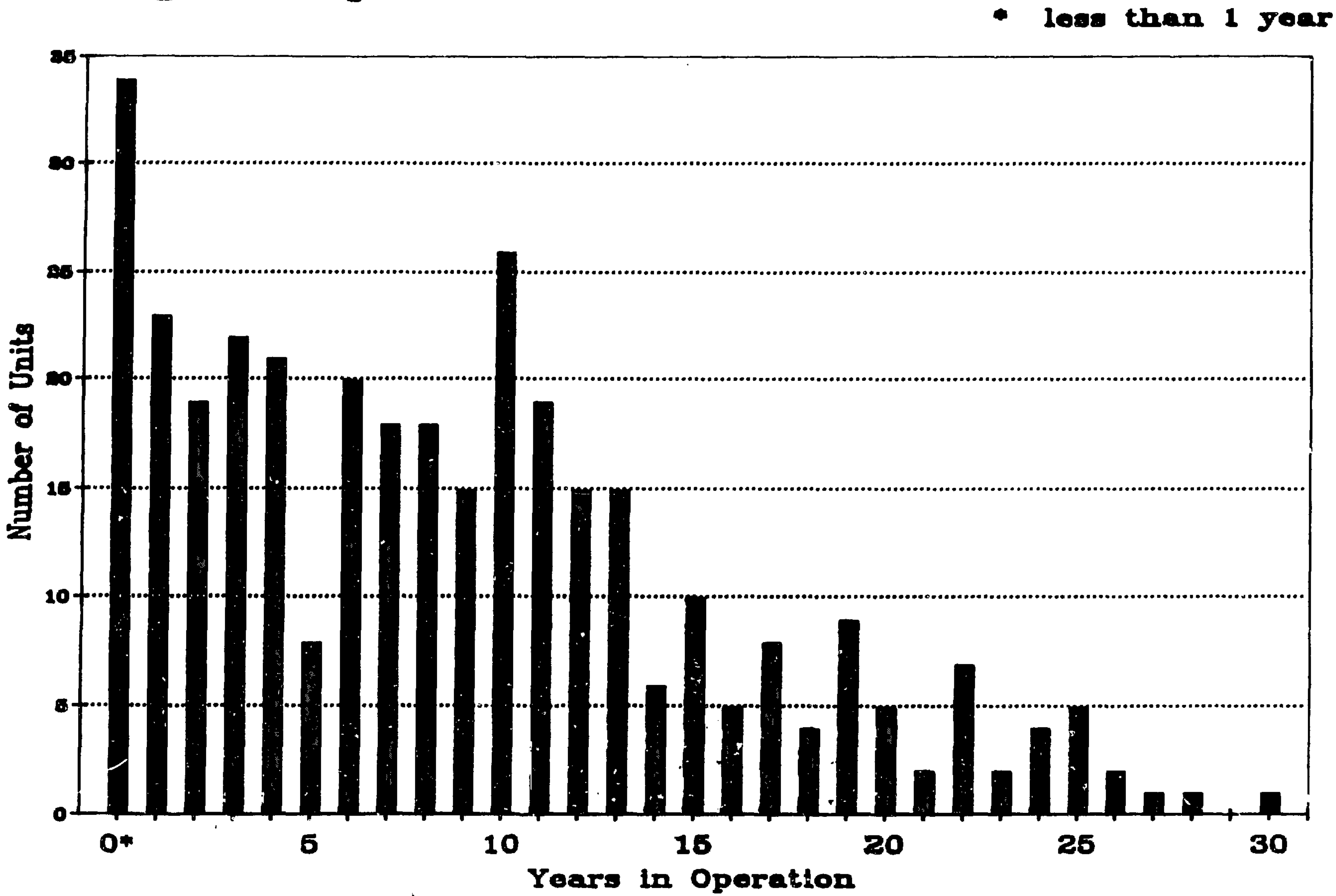
You can also come up with that number by looking at the cumulative operating experience in the nuclear power field. By the end of 1984, there were approximately 3,500 reactor-years of operation. There are about 345 reactors operating, so if you divide the two numbers, you come

OPERATING COMMERCIAL NUCLEAR POWER PLANTS

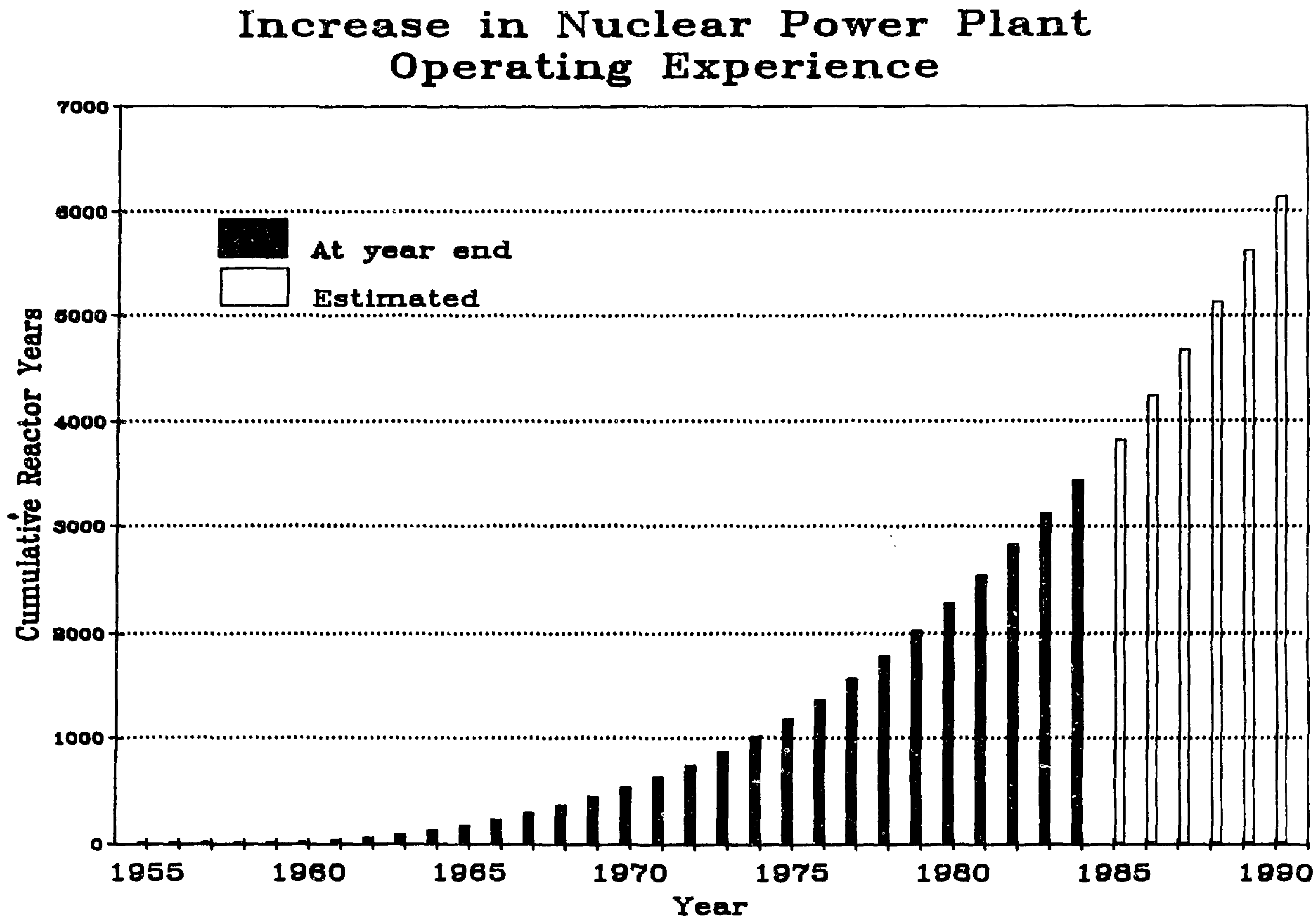
COUNTRY	NUMBER OF UNITS: LESS THAN 5 YEARS	5 TO 10 YEARS	10 TO 15 YEARS	MORE THAN 15 YEARS	MWe OUTPUT
ARGENTINA	1	--	1	--	935
BELGIUM	2	2	1	1	3 473
BRAZIL	1	--	--	--	626
BULGARIA	2	1	1	--	1 632
CANADA	7	4	4	1	9 521
CZECHOSLOVAKIA	2	1	--	--	1 194
FINLAND	2	2	--	--	2 310
FRANCE	27	6	3	5	32 993
GERMAN Dem. Rep.	--	2	2	1	1 694
GERMANY, Fed.Rep.of	6	7	3	3	16 114
HUNGARY	2	--	--	--	805
INDIA	2	--	1	2	1 020
ITALY	--	1	--	2	1 286
JAPAN	9	12	8	2	21 751
KOREA, Rep.	2	1	--	--	1 790
NETHERLANDS	--	--	1	1	508
PAKISTAN	--	--	1	--	125
S. AFRICA	1	--	--	--	921
SPAIN	4	--	2	1	4 690
SWEDEN	4	2	4	--	7 355
			2		
SWITZERLAND	1	1		1	2 882
TAIWAN	3	2	--	--	4 011
UNITED KINGDOM	5	5	2	25	9 564
U.S.A	19	21	37	8	68 867
U.S.S.R.	16	9	8	13	22 997
YUGOSLAVIA	1	--	--	--	632
	119	79	81	66	219 696

December 1984

Age of Operational Nuclear Power Plants

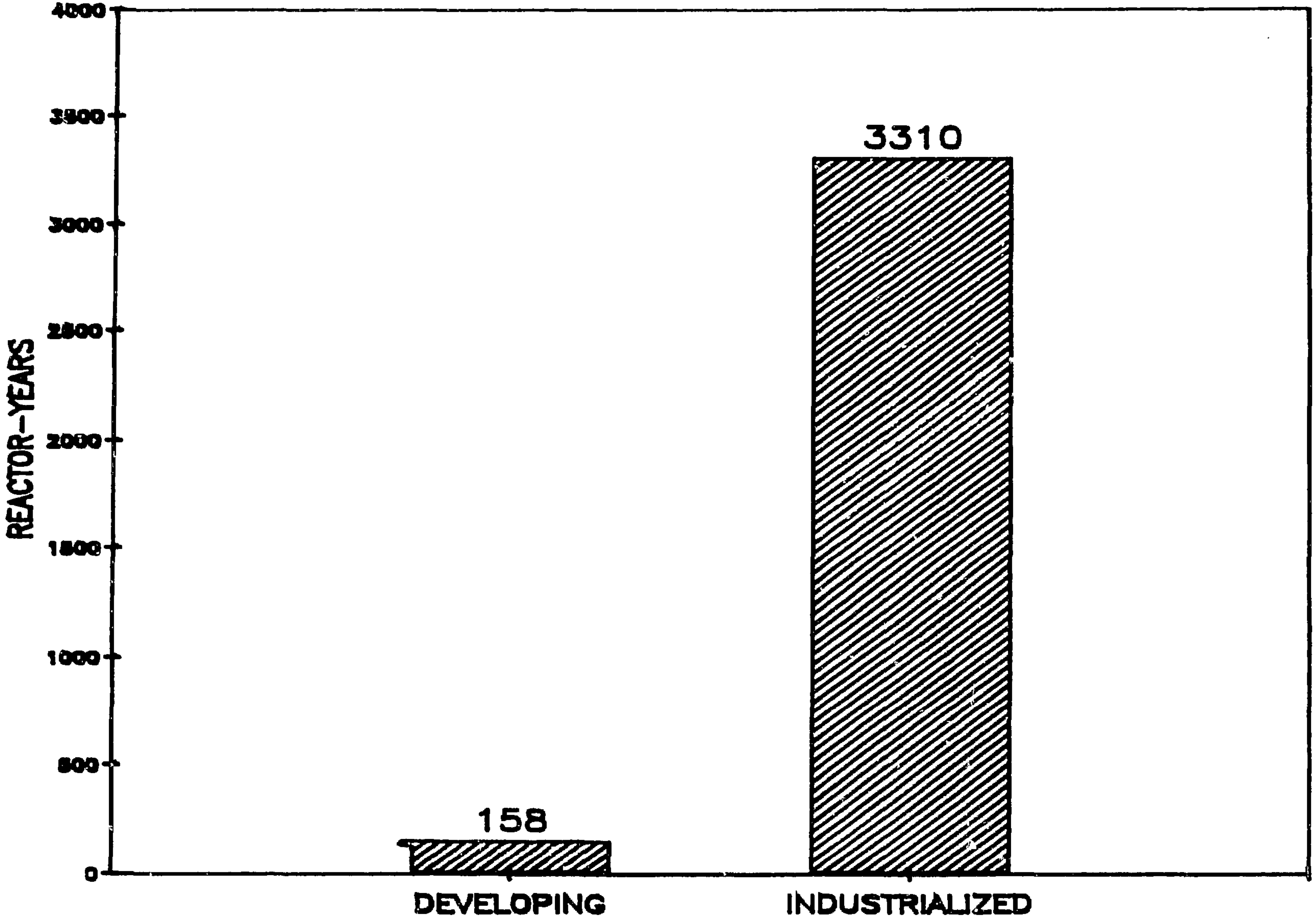


Source: IAEA Power Reactor Information System (PRIS)



Source: IAEA Power Reactor Information System (PRIS)

RELATIVE OPERATING EXPERIENCE



NUCLEAR POWER PLANTS IN DEVELOPING COUNTRIES

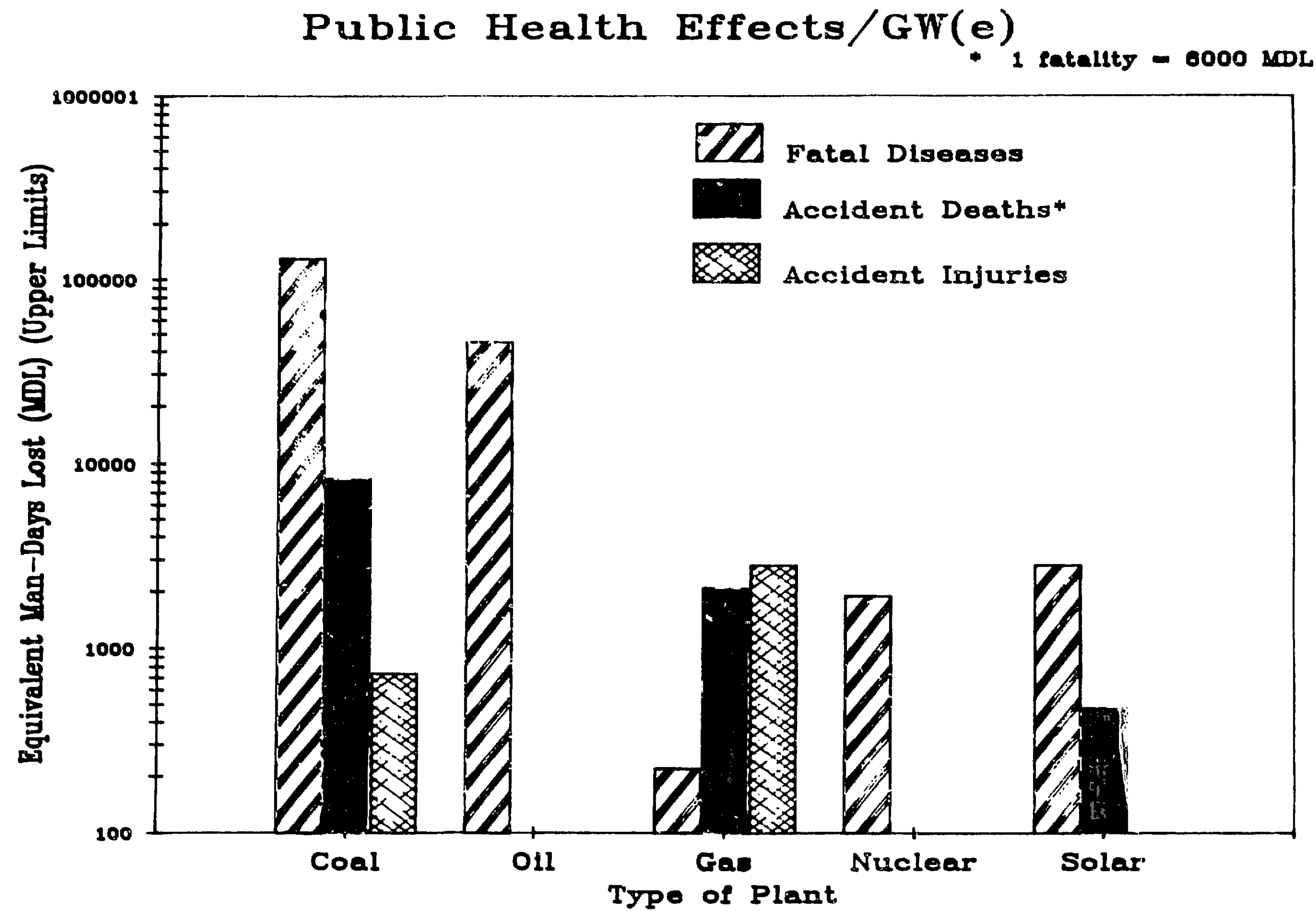
COUNTRY	1983	1984	1985*	1986*	1987*
ARGENTINA	2	2	2	2	2
BRAZIL	1	1	1	1	2
BULGARIA	4	4	4	5	5
CUBA	–	–	–	–	1
CSSR	2	4	6	8	9
HUNGARY	1	2	3	4	4
INDIA	5	5	6	6	7
KOREA, REPUBLIC OF	3	3	4	6	7
MEXICO	–	–	1	2	2
PAKISTAN	1	1	1	1	1
PHILIPPINES	–	–	1	1	1
ROMANIA	–	–	–	1	1
YUGOSLAVIA	1	1	1	1	1
TOTALS	20	23	30	38	43

* ESTIMATES

to 10 years as an average age for the operating reactors. Those numbers of operating experience will double in about another 7 years, so that we will be up to about 7,000 reactor years of operation, so that we are accumulating operating experience rather rapidly.

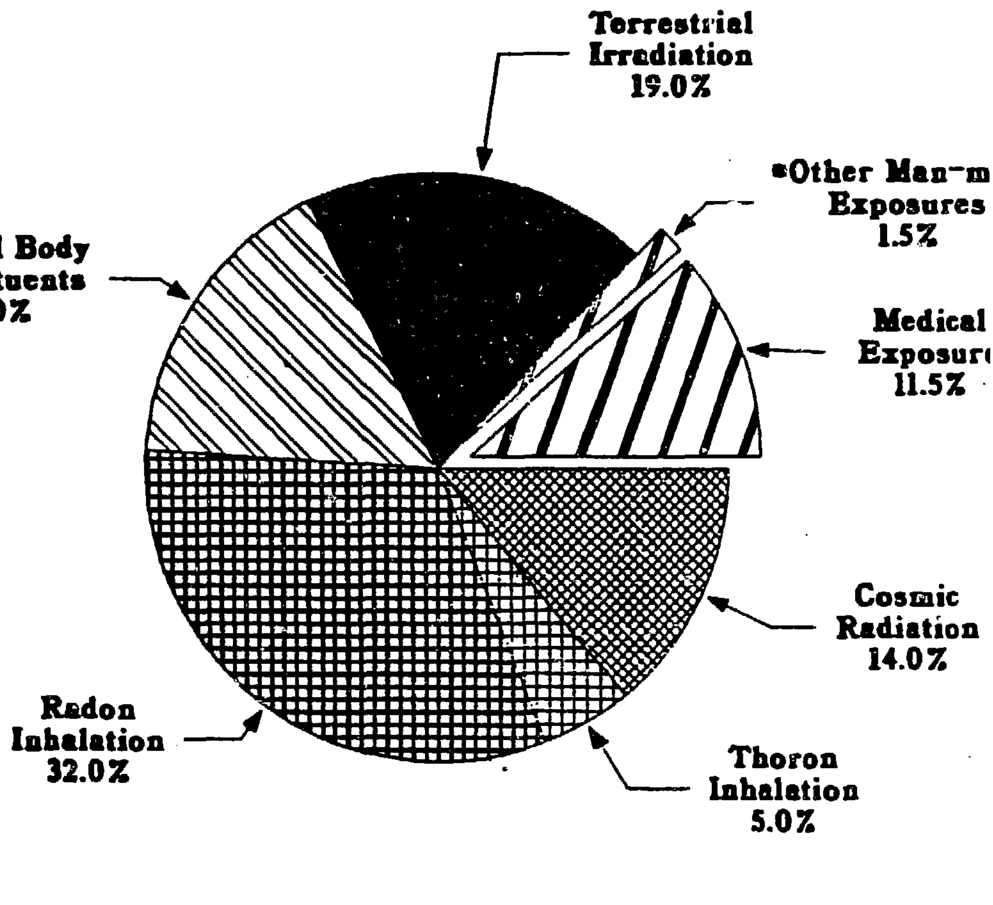
One point that should be made is that most of this operating experience is coming from the operating reactors in the developed countries. If you look at the breakdown in the next chart, you will see that, out of the approximate 3,500 reactor years of operation, about 3,300 come from the developed countries and about 158 reactor years of operation come from the developing, and the experience is different. There are things to be learned from the developing countries. They have unique situations. The grid networks are normally not the same. The maintenance and the supply of spare parts may not be the same, so that the operating history and experience from the two groups, the developed and the developing countries, will be somewhat different. Finally, on this overview of the world-wide situation, let me show you the last chart, which indicates the nuclear power plants in the developing countries. It shows 13 developing countries, with about 43 reactors projected to be operating in 1987.

The agency has been interested in some of the comparative work with energy sources. It's a very difficult area to talk about, when you try to compare coal and oil and solar and nuclear. The data available are not complete. There's much interpretation that has to go into it. But we have done some work just to put some perspective on where nuclear stands as far as public health hazards and as far as occupational health hazards. And I'll just show you one study that we did on the public health effects of nuclear power. It was for a 1,000-megawatt reactor, and it was to indicate the health effects, fatal diseases, accidents, and accidental injuries due to coal, oil, gas, nuclear, and solar. We used man-days of lost time, where a death was 6,000 man-days lost, and then summed them up for the various energy sources. Again, there's a lot of interpretation and a lot of assumptions that go into these data, so I'm not trying to defend them. But they do, surprisingly to some people, indicate that solar has some health effects to the public, normally



Source: Symposium on Environment and Industrial Health, 3–5 Feb 1985, New Delhi, India coming from the large transportational requirements, the large production requirements of solar cells. These figures that are indicated here include the production of the solar cells and the health effects due to the industry's producing it. Coal comes out to be on the high side. This would be equivalent as far as fatal diseases to about 20 deaths per year. On the nuclear side, it would come down to about one. Again, I just show it to you as a reference. There is some value to doing this type of work, especially to try to put risk into perspective for the public. But it's a difficult area, difficult to explain, and it's a dangerous area to get into unless you really know the subject and have the verbal skills to communicate it.

In another area, trying to show the effects of nuclear power, I'll show you one chart, which was a recent study that was done in the United Kingdom as far as radiation exposure. It doesn't have anything surprising, in a way, but it does graphically show that, if you talk about radiation exposures to the public, about 87% of it is natural, and you can do very little about it. A lot of the natural is coming from radon inhalation and thoron inhalation from building materials and soil. It's an area of growing interest as far as radiation exposure. In addition, there's the cosmic radiation, the natural body constituents, potassium-40, and then some of the radiation just coming from the soil. And you find out in this United Kingdom study that only 13% was man-made, and the medical is the major part of that, about 11.5%. Only 1.5% comes from other man-made sources, which includes the nuclear. It turns out, in the United Kingdom, only about 0.1% is coming from the nuclear plants. An additional 0.1% is coming from the fly-ash from coal-fired plants, so that in this study in the United Kingdom, the amount of radiation exposure from the fossil plants, because of the radioactive material that is contained in some of the coal, was equal to the radiation exposure coming from the nuclear plants. It also turns out that fall-out from the weapons testing of the '50s and early '60s is introducing 0.5% per year in the United Kingdom, which is about five times what comes from nuclear plants. Again, I give you these numbers to show you some of the recent studies on nuclear power, to put it in a little more perspective.



Terrestrial Irradiation 19.0%
*Other Man-m Exposures 1.5%
Medical Exposur 11.5%
Natural Body Constituents 17.0%
Cosmic Radiation 14.0%
Radon Inhalation 32.0%
Thoron Inhalation 5.0%

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So now let me get to the agencies' programs themselves. Some of these, again, you are familiar with, but I will introduce a few that perhaps will be new to you. One of the oldest areas in the safety work of the agency has to do with safety standards. We have three major programs in the safety standards area. The nuclear safety standards program is the biggest one. Many of you are familiar with that one. We also have the basic safety standards for radiation protection and the regulations for the safe transport of radioactive material. These three are the biggest and most important safety standards programs at the agency. You have probably seen the documents. The chart that you are going to see now is in color. You'll see it in black and white, but you're familiar with this red and yellow make-up of the nuclear safety standards, the ones that govern the nuclear power plants. The coding system at the agency uses red for a safety standard, and normally the top is white. But for the the Nuclear Safety Standards Program, we put a particular yellow. But normally, if you see a document at the agency which is red, it's a safety standard. On this particular chart, you'll see the five major groupings of the nuclear power safety standards. There's a standard on governmental organizations, the one on siting, one on operation, one on quality assurance, and one on design. Each one of these standards is backed up by about 10 safety guides, and the safety guides show how to implement these standards. The guides are always with the green color, so if you see a document at the agency that's green, you'll know it's a guide that is a back-up to the particular safety standard. Again, you'd normally see a white top, but for these particular nuclear safety standards programs, we use a yellow. I won't go through the particular titles of many of the guides on siting, for example, they'll talk about the site selection. There are guides on seismic, there are guides on earthquakes, hurricanes, etc. They're in quite a bit of detail. There's now about 2,000 pages contained in these safety standards and guides. They're published in the four official languages of the agency: English, Spanish, French, and Russian. In China, they are now translating the entire set into Chinese, the entire set being five standards and a total of 60 documents, if we include the guides. They will eventually be translated into Arabic, so that, in the

IAEA STANDARDS AND REGULATIONS

NUCLEAR SAFETY STANDARDS PROGRAMME

BASIC ~~SAFETY STANDARDS FOR RADIATION PROTECTION~~

REGULATIONS FOR THE SAFE TRANSPORT OF RADIOACTIVE MATERIALS long run, we will probably have these codes and guides in six languages. That program began in 1974. It is going to be completed at the end of this year. We have not come up with a definitive plan as to what we will do in the future, but in general we will make reviews on an ad hoc basis. We will review the documents and redo them only when we think there's a need in a particular area. Otherwise, these documents will stand for a long period of time. So, let me not go into the particular titles of these guides.

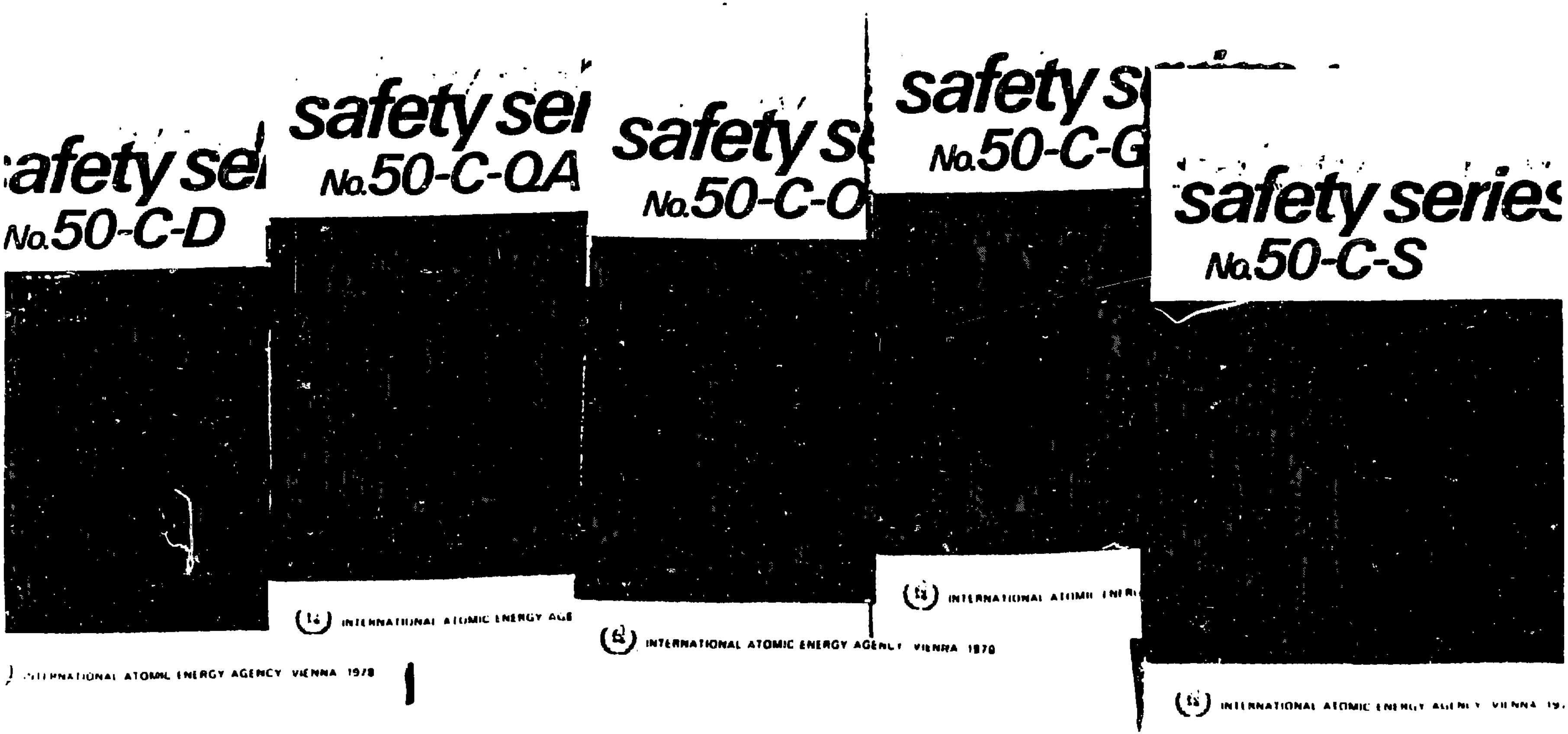
I'll just switch quickly to the other safety standards. I won't project this one because it's kind of dark and it won't project too well. It's Safety Series No. 9, which is the one on radiation protection, and it's called "The Basic Safety Standards for Radiation Protection." It was revised in 1982. It contains the latest ICRP recommendations. The ALARA principle is in here, and, of course, benefits analysis. So it's the incorporation of the latest radiation protection advice.

The third standard that I showed you on transportation is Safety Series No. 6. It's the standard on radioactive material transport. It's backed up with this green guide, and several other guides in the transport area. Again, this one now has a white top and the customary red for the standard and the green for the guide. Another standard that we've developed is on mining and milling of radioactive ores. It's Safety Series No. 26. There are a number of others that are at the agency now.

Let me stop now on the standards and turn to another area at the agency, which is the exchange of information. Again, many of you are familiar with that. We do this very simply by meetings. They are of various sizes. We run a lot of conferences. We run a number of symposia each year, and we have a number of seminars. The seminars are normally for training purposes, and much smaller. The conferences can be as high as a thousand participants, and the symposia perhaps two or three hundred. As a result of many of those meetings, we have produced a multitude of documents, and for every meeting we normally produce a document. So that at the agency you can get a catalogue which will describe the results of most of these meetings. Many are of particular value in the

NUclear Safety Standards

Five
CODES of
PRACTICE



nuclear power area. I'll show you one that we are now concentrating on, and that's mutual emergency assistance in case of an accident. We have produced a document which is called "Guidelines for Mututal Emergency Assistance". We are now also developing a number of backup documents in that particular area. It's not a simple area, when you talk about emergency assistance on an international basis. There are many constraints to emergency assistance. They can be simple political constraints, countries that are neighboring that do not have good political relations. They can be financial -- who pays in case of an incident that involves two countries? The legal liability. Commercial secrecy. There are other constraints just on customs and how do you get into a country. Much of this will be covered in these guidelines that we have developed and are continuing to develop.

Another area of information exchange -- we have just started the International Nuclear Safety Advisory Group. We use the word INSAG, the acronym, to describe it. The first meeting of this group was in March of this year, and we selected 13 individuals from a wide spectrum of countries which involved the developing countries and the socialist countries and the OECD countries. Of these 13 members, we have three from the developing countries, one from Korea and one from China and an individual from Brazil. From the socialist countries, we have the Soviet Union and the German Democratic Republic. This group has selected three topics to concentrate on; they are three pretty prominent ones in the nuclear area: the source term, feedback of operating information, basically called incident feedback, and the human elements area. What the committee has done now has been to appoint three individuals, one on each of the groups, to start organizing the work in this area. The group will not be giving regulatory advice. However, they hopefully will take some of these items and produce some definitive reports, in particular on the source term. On the source term, the agency will run, at the end of October, a symposium at Battelle Columbus, that will review much of the progress that has been made on the source term. The second meeting of this International Nuclear Safety Advisory Group will be in October of this year. Herb Kouts, who was at the early part of this meeting, is also in that group.

Rosen

INFCIRC/310

GUIDELINES FOR MUTUAL EMERGENCY ASSISTANCE ARRANGEMENTS IN CONNECTION WITH A NUCLEAR ACCIDENT OR RADIOLOGICAL EMERGENCY

INTERNATIONAL ATOMIC ENERGY AGENCY

I N S A G

International Nuclear Safety Advisory Group

INSAG

- *SOURCE TERM*
- *INCIDENT FEEDBACK*
- *HUMAN ELEMENTS AREA*

Another document that we are producing for information exchange is an Annual Nuclear Safety Review. The charter I have here is just the front page of the 1983 edition. We have a 1984 edition. It's a document of about 100 pages, and it gives the highlights, the status, the outlook on nuclear safety, worldwide. I think it's a document that is getting better each year. The one we have just produced will receive wide distribution, and it contains some factual information that I think could be of interest to many of you. They will be made available at this year's general conference.

Just a word about information exchanges. We will have a large conference about it in 1987, and it has to do with the area of human factors, operational safety, and it will be called the International Conference on Man/Machine Interface in the Nuclear Industry. Again, it will be run in 1987. We have not picked a location or set the agenda for the meeting yet.

Let me now go to the last guideline on information exchange and just mention the IRS system which we have started at the agency. That's an Incidental Recording System. It's a system that's not meant to compete with INPO or some of the larger systems. It's a more modest attempt to have an exchange of incident reports worldwide. It would cover the more significant incidents; we're talking about receiving on the average a half incident per reactor per year. So, for example, for the 345 operating reactors, if we received about 150 incident reports per year, we would at this stage be satisfied. It's an incident-reporting system, but perhaps more similar to the one that the International Civil Aviation Organization is using in the aircraft industry. It is looking for the more severe incidents, and is not meant to be a statistical accumulation of events. It's more to get an interchange started between developing countries and the OECD countries and the socialist countries. Of the 26 countries with nuclear power, almost all of them have officially joined the system. We hope that over the next few years you will become more familiar with it, and we will start collecting incidents that can be the basis of a number of meetings where we can discuss and analyze these incidents. We have produced some

NUCLEAR SAFETY REVIEW 1983

HIGHLIGHTS, STATUS AND OUTLOOK

1987

INTERNATIONAL CONFERENCE

ON

MAN–MACHINE INTERFACE

IN THE

NUCLEAR INDUSTRY

OP16

PROVIDING THE EXCHANGE OF INFORMATION:

IRS

documents that could be of use to you in that area. For example, we have a guide on how to establish a national system for collecting, assessing, and disseminating information on incidents. That document is basically the initial phase of our international system, because you really must have national systems similar in a number of countries before you can bring it together in some kind of international group. We have been working closely with the NEA in Paris. The NEA also has an incident-reporting system, and we hope in the future to bring the two systems together into one. The NEA, of course, represents the OECD countries, whereas the agency has access to the developing countries and the socialist countries.

I will now talk a little bit about our expert services, and how we perhaps can assist you in a variety of safety areas. I'll just put one brief picture of what we mean by advisory services at the agency. We simply mean to bring together a number of experts. We normally send them to a developing country on a specific topic, and they may stay a week or two weeks, and basically give advice. We give these advisory services in many areas. We have done it to help establish regulatory organizations. We have done it in the siting area. We have done it in the operations. The chart that you see here is really an old chart. I think it was done in 1981 or so, and it shows some typical countries and the advisory services that we have given to these countries. In this particular area of advisory services, I want to mention the new program that we have at the agency that is called the OSART -- the Operational Safety Review Teams. Its an acronym that you may become familiar with, we hope you will. It's similar to what INPO does in their reviews in the U.S. In fact, we have patterned these Operational Safety Review Teams in cooperation with INPO, and we have patterned them after the INPO type of review. The basic objectives of the Operational Safety Review Teams is an independent review for the regulator, in this case, so that is quite different from the INPO review. The INPO review is basically a utility review. But we are doing this in a way to support the national authorities in performing a review. It is not meant to be an inspection, however. In that respect it is more similar to INPO. It is meant to give

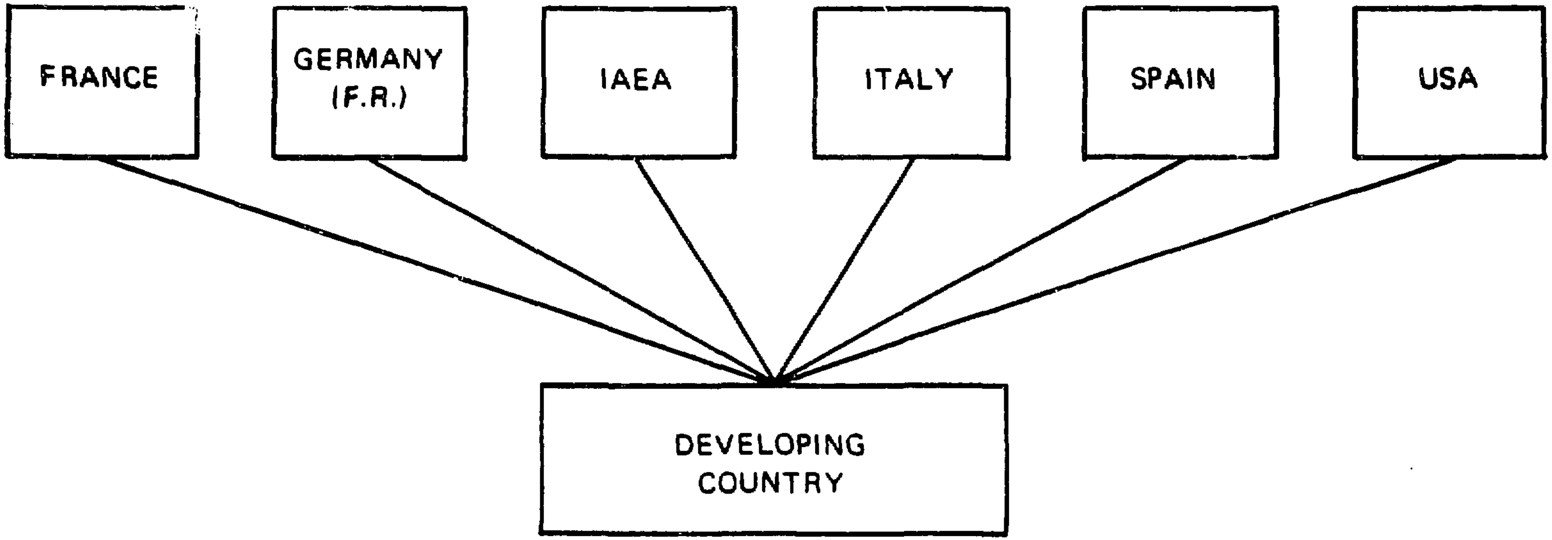
IAEA-TECDOC-278

GUIDE ON A NATIONAL SYSTEM FOR COLLECTING, ASSESSING AND DISSEMINATING INFORMATION ON SAFETY-RELATED EVENTS IN NUCLEAR POWER PLANTS

A PRE-PUBLICATION WORKING DOCUMENT

A TECHNICAL DOCUMENT ISSUED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1983

Mission of Experts



1980 Missions

EGYPT MOROCCO	Siting
BRAZIL SPAIN YUGOSLAVIA	Safety Aspects

Table 2. Nuclear power safety assistance to Member States (since 1975)*

Missions**			
Siting	Safety report review	Regulatory body advisory	Nuclear legislation advisory
Argentina	Brazil	Brazil	Algeria
Chile	Iran	Chile	Brazil
Egypt	Korea, Rep. of	Egypt	Egypt
Indonesia	Philippines	Greece	Kuwait
Kuwait	Yugoslavia	Korea, Rep. of	Libyan Arab Jamahiriya
Libyan Arab Jamahiriya		Mexico	Malaysia
Malaysia		Pakistan	Morocco
Morocco		Philippines	Yugoslavia
Pakistan		Portugal	
Peru		Spain	
Philippines		Syria	
Turkey		Turkey	
Venezuela			
Yugoslavia			

Expert assistance		
Long term***	Short term	
Brazil	Argentina	Korea, Rep. of
Korea, Rep. of	Brazil	Mexico
Mexico	Bulgaria	Philippines
Philippines	Chile	Portugal
Yugoslavia	Greece	Romania
	Iran	Turkey
	Israel	Yugoslavia

O S A R T

OPERATIONAL SAFETY REVIEW TEAM

OPERATIONAL SAFETY REVIEW TEAM (OSART)

OBJECTIVE:

INDEPENDENT REVIEW OF OPERATIONAL SAFETY AT THE NUCLEAR PLANT

SUPPORT FOR THE NATIONAL AUTHORITIES IN AUDITING AND VERIFYING OPERATIONAL SAFETY

ASSISTANCE TO THE OPERATING ORGANIZATIONS BY MAKING AVAILABLE EXPERIENCE FROM COMPARABLE PLANTS

DIRECT CONTACT WITH NATIONAL AND INTERNATIONAL EXPERTS

NOT A SUBSTITUTE FOR REGULATORY SAFETY EVALUATION advice, it's not meant to be a substitute for a safety review or a regulatory-type review. The reviews are somewhat similar to INPO in its composition -- we also use eight to twelve experts. Some of the experts come from the agency; the experts coming from the agency are normally associated with the topics that are similar for the variety of plans, whether it be the BWR or PWR, for example, in training, in emergency preparedness, an agency expert is normally sent. In the more specific operational areas, we will try to bring in someone who is an expert in that particular type of reactor, so for a PWR from Westinghouse we may bring in people familiar with the Westinghouse-type PWR. These are not cheap reviews, and for funding we normally require the developing country to pay the local expenses. The local expenses can normally run to about $10,000 per review. That's normally just for per diems and local transportation. The agency pays the rest of the expenses, and the rest of these expenses means transportation, bringing in the experts. Normally, we do not pay salaries to the experts, but there are cases where we do have consultants who are retired or do not have a direct source of income, and in those cases we pay. In these operational safety reviews, we are also bringing in observers, similar to INPO, and many of these observers will come from developing countries, so that they will get first-hand experience in conducting an operational safety review. The areas we cover include organizational administration, training and qualification, operations, maintenance, etc.

Here is a list of a few of the reviews that we have done, and I'll show you the composition of one of the reviews as a closing to that particular area. We started our first one in Korea. We have also done one in Yugoslavia. In the Philippines we did what we would call a pre-operation review, because in the Philippines the plant is not under operation. We have gone to Pakistan. We did another one in the Philippines, and we are presently in Brazil this week and next week. The date you see here has been postponed. We will do one in France in October. That will be more a training exercise, but the French hope to set up a body that will follow these particular operational safety reviews and bring a core of international experience into France. We

OPERATIONAL SAFETY REVIEW TEAM
(OSART)

COMPOSITION:

EIGHT TO TWELVE EXPERTS WITH LONG TERM NUCLEAR EXPERIENCE

IN–HOUSE MEMBERS FOR CONSISTENCY AND FOR AREAS WHICH DO NOT DIFFER AMONG VARIOUS REACTOR TYPES

EXTERNAL CONSULTANTS FOR AREAS WHICH DIFFER AMONG VARIOUS REACTOR TYPES

COST–FREE OBSERVERS FOR TRAINING PURPOSES

FUNDING:

- LOCAL EXPENSES BORNE BY RECEIVING COUNTRY
- OTHER EXPENSES ARE COVERED PRIMARILY WITH TECHNICAL COOPERATION FUNDS (INTER–REGIONAL AND NATIONAL)

OBJECTIVES OF PLANT EVALUATION

ORGANIZATION AND ADMINISTRATION

TRAINING AND QUALIFICATION

OPERATIONS

MAINTENANCE

RADIATION PROTECTION AND CHEMISTRY

EMERGENCY PREPAREDNESS

TECHNICAL SUPPORT

OSART REVIEWS PERFORMED AND PLANNED

PERFORMED :

KOREA, REPUBLIC OF – KO-RI UNIT 1 .. 08–26 AUGUST 1983

YUGOSLAVIA – KRSKO .. .. 06–24 FEBRUARY 1984

PHILIPPINES – PNPP, UNIT 1 .. .. 25 JUNE TO 12 JULY 1984

PAKISTAN – KANUPP .. .. 07–20 JANUARY 1985

PHILIPPINES – PNPP, UNIT 1 .. .. .. 04–15 FEBRUARY 1985

BRAZIL – ANGRA, UNIT 1 .. .. 24 APRIL TO 14 MAY 1985

PLANNED :

FRANCE – PLANT TO BE SELECTED

KOREA, REPUBLIC OF – KO-RI, UNIT 2
WOLSUNG, UNIT 3

SPAIN – PLANT TO BE SELECTED

MEXICO – LAGUNA VERDE, UNIT 1

hope to do one in Spain. Mexico, I think we have scheduled for the early part of next year, January or February. We are working on doing one in Finland. Finland has a Soviet-type reactor. One of the reactors is Soviet-designed, and we hope by doing one in Finland, if we can do it with the Soviet-type-design reactor, then we can perhaps expand into the socialist countries. So it's a program that we think will grow in the years to come, and that's why I mentioned at the beginning a shift towards the operational work at the agency, including a shift of manpower towards the operational part.

I'll indicate one other area in radiation protection that we are introducing, not similar to the OSART; we are going to call it by the acronym RAPAT, which means Radiation Protection Advisory Team. We have found in many of the developing countries that one of the weak points is in radiation protection. If you look at the significant accidents that have occurred in the past several years, you find that many of them are in the area of radiation protection, and not in the area of nuclear power plants. Some of these you're familiar with; with some you may not be. One occurred rather recently in Morocco leading to the deaths of eight people from a radiation source. That's unfortunately not becoming a unique event. So that in this area the agency is beginning to concentrate, and we will send out these advisory teams. The teams are much smaller and have a different purpose from the OSART. The teams here are normally three or four experts. They can in fact be two, and the review normally lasts three to ten days. The objective is to assess the current status of radiation protection in the country and to develop some long-term assistance plans, including agency contributions. We have done a number of these reviews and it appears that they will be very popular. We started in China at the end of last year and we have been to Iraq, Nicaragua, Syria, Paraguay. We have a number of other requests, which are listed on this slide.

Let me finish by talking a little more about technical assistance. This figure shows the increasing amount of technical assistance coming from the agency. It shows the assistance through the years. You can see the growth in nuclear safety in the last few years and that area is

RAPAT

R**A**DIATION **P**ROTECTION **A**DVISORY **T**EAM

RAPAT

DURATION: 3 – 10 DAYS

COMPOSITION: 3 – 4 EXPERTS

OBJECTIVE: ASSESS CURRENT STATUS
DEFINE IMMEDIATE and FUTURE NEEDS
FORMULATE LONG–TERM AGENCY ASSISTANCE

RAPAT

PERFORMED: CHINA (NOV 1984), IRAQ (DEC 1984), NICARAGUA (FEB 1984), SYRIA (FEB 1985), PARAGUAY (FEB 1985)

REQUESTED: CHILE, ECUADOR, GABON, JORDAN, MALAYSIA, MEXICO, MOROCCO, PAKISTAN, PHILIPPINES, THAILAND

OP18

growing again. Technical assistance sponsors much of what we do in the Safety Division, in particular, training courses. All of our training courses are from technical assistance funds. This chart indicates some of the training courses in nuclear power safety that we have run. In that area, we run about six courses a year, in various aspects of nuclear safety and radiation protection. We've done it in quality assurance, siting, operations, maintenance, and the whole spectrum of nuclear power. The courses normally consist of about 30 participants. It's a training course, let's say, similar to what you have here, normally double the size of this one, and they run sometimes for as little as three weeks, but in most cases more. We have run them up to about ten weeks in the radiation protection area. In fact we have run them for six months. In the radiation protection area, we have a training course in Argentina which has been run for many years. We started one in India this year, in the English language, for six months. It was actually a little shorter, about five months. And we plan to expand that to French next year, and to Russian. So, training is a big part of the activity at the agency and the technical assistance funds pay for the training. Technical assistance also pays for some of the advisory services we run, and technical assistance I think in the case of some of you has paid for your trip here as part of a scientific visit. Technical assistance is also used for scientific visits. If you find it worthwhile to visit certain installations, including operating reactors, if we can get permission for you to visit, we can set up scientific visits that can last for two or three weeks at a number of installations. That, again, comes from technical assistance. I just want to repeat that it's a big part of the effort at the agency. It supplies in the safety area about half of our funds.

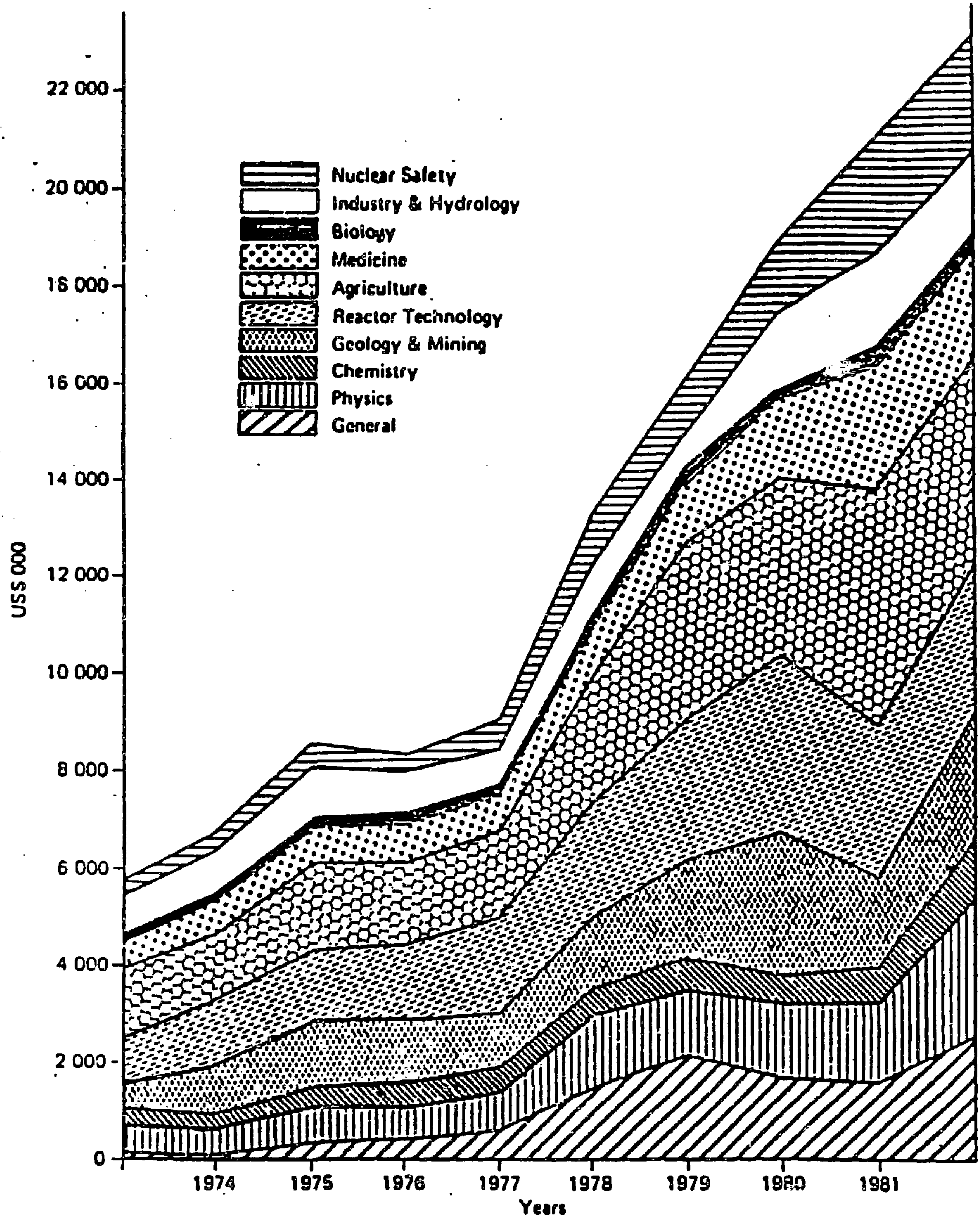


Figure 1.
IAEA Technical
co-operation programme
by field of activity:

Table 3. IAEA Inter-regional safety-related training courses (1978–1983)

Course*	Location	Starting date		Duration (weeks)
Safety Analysis Review	Argonne (USA)	Aug.	1978	8
Quality Assurance	Argonne (USA)	Oct.	1978	5
Siting for Nuclear Power Plants	Argonne (USA)	Sep.	1979	9
Quality Assurance	Madrid (Spain)**	Oct.	1979	6
Safety and Reliability in Operation	Argonne (USA)	Nov.	1979	6
Safety Analysis Review	Karlsruhe (FRG)	Nov.	1979	4
Environmental Impact Assessment of Nuclear Power Plants	Argonne (USA)	Mar.	1980	6
Inspection of Nuclear Power Plant Construction	Argonne (USA)	Jun.	1980	9
Safety Analysis	Karlsruhe (FRG)	Sep.	1980	6
Regulation of Nuclear Power Plants	Argonne (USA)	Sep.	1980	9
Quality Assurance	Karlsruhe (FRG)	Oct.	1980	6
Safety Analysis Review	Argonne (USA)	Mar.	1981	8
Radiation Protection and Nuclear Safety	Buenos Aires (Argentina)**	Jun.	1981	7
Operational Safety	Karlsruhe (FRG)	Sep.	1981	6
Siting	Argonne (USA)	Sep.	1981	7
Radiological Emergencies Planning	Argonne (USA)	Feb.	1982	3
Seismic Considerations in Siting	Argonne (USA)	Feb.	1982	5
Risk Prevention	Saclay (France)***	May	1982	4
Siting	Saclay (France)***	Oct.	1982	4
Quality Assurance	Saclay (France)***	Apr.	1983	5
Probabilistic Risk Assessment	Argonne (USA)	Sep.	1983	4

* About 30 participants per course.

** Conducted in Spanish.

*** Conducted in French