CSNI Report No. 43
Volume 3

SPECIALIST MEETING
ON
REGULATORY REVIEW IN
THE LICENSING PROCESS

Madrid, Spain

7-9 November 1979

Hosted by the

JUNTA DE ENERGIA NUCLEAR

PROCEEDINGS

April 1980

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EXAMENS RÉGLEMENTAIRES DE SÛRETÉ ET ÉTUDES DE SÛRETÉ

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Les risques potentiels associés à la production d'énergie nucléaire ont conduit les pouvoirs publics français à mettre en place un système spécifique d'autorisations ; schématiquement, les installations nucléaires importantes font l'objet de trois autorisations successives importantes : l'autorisation de création - qui doit normalement précéder tous travaux de caractère irréversible sur le site -, l'autorisation d'essais de mise en service et l'autorisation de mise en exploitation normale - qui suit les essais de mise en service et le début d'exploitation de l'installation. Pour obtenir chacune de ces autorisations, l'exploitant d'une installation doit soumettre un rapport de sûreté (il s'agit respectivement du rapport préliminaire de sûreté, du rapport provisoire de sûreté et du rapport définitif de sûreté, les deux derniers étant accompagnés de propositions de règles générales d'exploitation) où il précise les dispositions techniques et d'organisation qu'il a prises ou prévues pour assurer la sûreté de l'installation.

Ces rapports sont systématiquement examinés à la demande des autorités réglementaires françaises par l'Institut de Protection et de Sûreté Nucléaire du Commissariat à l'Energie Atomique qui rapporte les résultats de cet examen devant les groupes permanents d'experts chargés de donner un avis à ces autorités. Il y a donc, pour chaque installation importante, trois évaluations complètes successives de la sûreté de cette installation à trois stades différents de sa vie (non compris la mise à l'arrêt définitif).

L'analyse de la sûreté d'une installation nucléaire - qui peut être définie comme englobant la totalité des actions qui permettent d'évaluer les risques potentiels liés au fonctionnement de cette installation, d'apprécier la validité et l'efficacité des dispositions prises ou prévues pour réduire ces risques et de formuler en définitive un jugement sur la nature et l'ampleur des risques résiduels - recouvre en fait des actions menées par différents organismes. Il est clair en effet qu'il appartient d'abord à l'exploitant d'une installation nucléaire de présenter, dans ses rapports de sûreté, des démonstrations convaincantes visant à prouver que son installation peut être créée et exploitée sans risques inacceptables ;

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mais, par ailleurs, les autorités règlementaires françaises font procéder à un examen critique des démonstrations présentées par l'exploitant par un organisme indépendant, l'Institut de Protection et de Sûreté Nucléaire du Commissariat à l'Energie Atomique et cet examen est une part importante de l'analyse de la sûreté d'une installation.

Il faut remarquer par ailleurs que, si trois étapes importantes sont prévues auxquelles correspondent trois examens approfondis, l'analyse de sûreté d'une installation est en fait un processus pratiquement continu qui commence dès le choix des options importantes au stade de l'avant-projet, se poursuit tout au long de la conception, de la réalisation, des essais de mise en service, de l'exploitation et jusqu'aux opérations de mise à l'arrêt définitif. Diverses dispositions sont mises en oeuvre par les autorités règlementaires françaises pour assurer un examen critique des dispositions envisagées par les exploitants à tous les stades de la vie des installations ; ces examens critiques sont, là encore, effectués par l'Institut de Protection et de Sûreté Nucléaire du Commissariat à l'Energie Atomique qui, soit donne directement son avis aux autorités règlementaires françaises, soit, à la demande de ces autorités, rapporte devant les groupes permanents d'experts précités qui donnent à leur tour un avis. Les autorités règlementaires donnent à ces avis les suites qu'elles jugent appropriées notamment par le biais d'un système d'autorisations particulières. On retrouve donc, tout au long de la vie de chaque installation, des analyses de sûreté comportant les deux aspects soulignés ci-dessus - présentations de démonstrations par les exploitants et examens critiques par l'Institut de Protection et de Sûreté Nucléaire du Commissariat à l'Energie Atomique - ; il faut souligner à cet égard que l'établissement de relations suivies et confiantes entre les exploitants et les techniciens de cet institut apparaît comme une condition importante à la réalisation d'une analyse continue de la sûreté satisfaisante pour les deux parties, étant entendu que les autorités règlementaires gardent à chaque instant leur pouvoir et leur liberté de décision.

Le travail de l'Institut de Protection et de Sûreté Nucléaire du Commissariat à l'Energie Atomique, soutien technique des autorités règlementaires françaises en matière d'analyse de la sûreté des installations nucléaires, constitue ce qu'il est convenu d'appeler les examens règlementaires de sûreté.

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Les personnes chargées de mener à bien ce travail au sein de cet institut peuvent être schématiquement regroupées en deux catégories distinctes et complémentaires, les généralistes et les spécialistes. Les généralistes, groupés par types d'installation (centrales nucléaires à eau ordinaire, réacteurs à neutrons rapides, réacteurs de recherche, usines de séparation des isotopes de l'uranium, usines de traitement des combustibles irradiés,...) sont responsables de la bonne exécution des examens règlementaires de la sûreté des installations dont ils ont la charge, à tous les stades de la vie de ces installations ; à ce titre, il leur appartient de maintenir, comme indiqué ci-dessus, des relations aussi suivies que possibles avec les exploitants de ces installations. Leur expérience et leurs connaissances des problèmes techniques du type d'installation concerné, aussi approfondies soient-elles, ne leur permettent pas, en général, de donner seuls un avis sur les questions de sûreté soulevées ; ils font alors appel aux spécialistes des différentes disciplines techniques, telles que la mécanique, la métallurgie, le génie civil, le contrôle-commande,.., qui leur fournissent des réponses motivées tenant compte de l'état des connaissances dans ces disciplines. Il faut insister ici sur le fait que les analyses de sûreté, et en particulier les examens règlementaires de sûreté, sont très généralement pluridisciplinaires et il est clair qu'un dialogue constant entre généralistes et spécialistes est nécessaire pour la bonne exécution du travail de l'Institut de Protection et de Sûreté Nucléaire du Commissariat à l'Energie Atomique ; il faut ajouter encore que le fait de conserver, tout en l'individualisant clairement, l'Institut de Protection et de Sûreté Nucléaire au sein du Commissariat à l'Energie Atomique permet à cet institut de bénéficier d'un accès privilégié aux connaissances de cet organisme qui poursuit par ailleurs d'importantes missions en matière de recherche et de développement, sans nuire à l'indépendance nécessaire entre l'organisme chargé des examens règlementaires de sûreté et le Commissariat à l'Energie Atomique, exploitant d'installations nucléaires.

A ce stade, on voit apparaître une source d'études et essais de sûreté liée directement aux examens règlementaires de sûreté ; les réponses des spécialistes peuvent en effet comporter le constat de lacunes ou d'insuffisances qu'il peut apparaître nécessaire ou opportun de combler par des études ou des essais appropriés. L'Institut de Protection et de Sûreté Nucléaire du Commissariat à l'Energie Atomique peut alors être amené à promouvoir et prendre à sa charge certaines études ou certains essais.

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Un exemple d'action de ce type, entreprise dans le cadre de l'examen du rapport provisoire de sûreté de la centrale nucléaire de Gravelines, concerne la tenue de cette centrale aux ondes de choc pouvant résulter d'explosions dans son environnement industriel ; compte tenu de la situation très particulière de cette centrale (proximité du port de Dunkerque), les autorités règlementaires de sûreté françaises ont été amenées dès sa création à mettre en place des dispositions propres à empêcher l'implantation d'industries par trop dangereuses à proximité de la centrale et à exiger un dimensionnement particulier des installations (tenue à une onde de choc incidente triangulaire à front raide d'amplitude maximale 200 millibars et de durée 400 millisecondes) ; lors de l'analyse du rapport provisoire de sûreté de cette centrale, l'Institut de Protection et de Sûreté Nucléaire du Commissariat à l'Energie Atomique a estimé nécessaire, compte tenu de la complexité du problème et de l'utilisation de codes de calcul insuffisamment validés par l'expérience, de prévoir des essais sur maquette visant à vérifier la tenue globale de la centrale aux explosions et à apprécier les marges de sécurité.

L'Institut de Protection et de Sûreté Nucléaire du Commissariat à l'Energie Atomique peut également estimer nécessaire de procéder ou de faire procéder à des calculs permettant de vérifier par d'autres méthodes les résultats présentés par les exploitants ; il est donc directement intéressé au développement de moyens de calcul appropriés - il s'agira le plus souvent de moyens de calcul développés par d'autres unités du Commissariat à l'Energie Atomique, tels que le système CEA/SEMT élaboré par le Département d'Etudes Mécaniques et Thermiques du Commissariat à l'Energie Atomique pour l'analyse mécanique des structures des réacteurs nucléaires ; ce système a été notamment utilisé à la demande de l'Institut de Protection et de Sûreté Nucléaire pour procéder à une estimation des marges de sécurité à l'égard des risques de déformation excessive et d'instabilité plastique dans un piquage de tubulure d'un réacteur à eau pressurisée et pour procéder à une estimation des marges de sécurité par rapport à la rupture par survitesse d'un volant de pompe primaire d'un réacteur à eau pressurisée en supposant une fissure semi-circulaire dans la rainure du clavetage (l'importance et l'intérêt de ce dernier calcul sont évidents car il s'agit d'apprécier la position défendue par l'exploitant visant à faire admettre qu'il n'y a pas lieu de tenir compte de la possibilité de rupture d'un tel volant par survitesse).

L'institut de Protection et de Sûreté Nucléaire du Commissariat à l'Energie Atomique peut également se retourner, en cas de lacunes ou d'insuffisances constatées lors des examens règlementaires de sûreté, vers l'exploitant en lui demandant de mieux étayer ses démonstrations par des études ou essais complémentaires;

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c'est ainsi qu'avant le premier chargement du combustible dans la première tranche de la centrale nucléaire de Fessenheim, il a été demandé à Electricté de France de procéder à un essai d'endurance sur un élément combustible, compte tenu de la nouveauté à l'époque du combustible utilisé (combustible 17 x 17) et des difficultés d'extrapolation des données acquises sur les éléments combustibles déjà utilisés dans des réacteurs analogues (combustible 15 x 15) ; un essai de 3000 heures a alors été réalisé dans la boucle Super-Bec du Commissariat à l'Energie Atomique à Cadarache et a permis de vérifier le bon comportement vibratoire des éléments combustibles du type 17 x 17 au débit nominal et à la température nominale de fonctionnement prévus pour la centrale nucléaire de Fessenheim.

Bien entendu, les cas d'études et essais liés aussi directement à des examens règlementaires de sûreté sont relativement limités et le plus souvent ponctuels car l'exploitant d'une installation nucléaire importante, fondamentalement responsable de la sûreté de cette installation et de la démonstration de cette sûreté, sera amené de lui-même à procéder ou faire procéder autant que nécessaire à des essais de sûreté (ou, au stade du rapport préliminaire de sûreté, à prévoir un programme d'études et d'essais) afin d'apporter des réponses convaincantes aux questions de sûreté qui seront, bien entendu, abordées dans le cadre des examens règlementaires de sûreté ; si de telles démonstrations ne sont pas amenées sur certains points, les autorités règlementaires seront conduites à imposer certaines limites de fonctionnement aux installations examinées. A titre d'exemple, Electricité de France envisage à l'heure actuelle un fonctionnement de ses centrales nucléaires à eau pressurisée en téléréglage, ce qui conduirait pour les éléments combustibles de ces centrales à des variations cycliques de puissance à fréquence élevée dans une gamme de puissance de l'ordre de 10 % ; en l'absence de démonstration convaincante de la tenue des éléments combustibles dans ces conditions, ce type de fonctionnement est actuellement interdit par les autorités règlementaires françaises et des essais de qualification des éléments combustibles à l'égard de ce type de fonctionnement ont été entrepris dans le réacteur expérimental dénommé CAP (Chaufferie Avancée Prototype) implanté à Cadarache.

D'une manière générale, l'Institut de Protection et de Sûreté Nucléaire du Commissariat à l'Energie Atomique se place toujours, dans le cadre des examens règlementaires de sûreté, du côté de la prudence compte tenu de l'état des connaissances ; .../...

en ce sens, il n'y a pas de contradiction entre le fait d'accepter la mise en service d'installations nucléaires - et le programme nucléaire français est particulièrement important - et le fait de poursuivre de nombreuses études et de nombreux essais en matière de sûreté. Les marges de sécurité importantes qui sont quelquefois prises dans certains domaines où les données manquent pourront être ultérieurement abandonnées, à la lumière de résultats convaincants ; une telle attitude conduit par exemple à maintenir la prise en compte d'un accident global de fusion du coeur pour le dimensionnement de la centrale nucléaire à neutrons rapides de Creys-Malville alors qu'une évolution de la philosophie en la matière pourra peut-être apparaître acceptable pour les centrales nucléaires à neutrons rapides suivantes, à la lumière des résultats des études et essais effectués depuis la conception de la centrale nucléaire de Creys-Malville.

Il faut ajouter ici que, malgré cette approche générale prudente, il n'est pas possible d'exclure totalement que des études ou essais entrepris par exemple pour mieux apprécier des marges de sécurité fassent en fait apparaître des marges nettement plus faibles que l'estimation qu'auraient pu en donner les spécialistes ; l'expérience d'exploitation, qui constitue un ensemble particulièrement important d'esais en vraie grandeur pourra également faire apparaître que certaines questions ont été insuffisamment examinées ou apporter plus généralement des éléments nouveaux d'appréciation ; dans tous les cas, les données nouvelles seront introduites au fur et à mesure dans les examens règlementaires de sûreté et pourront conduire à des études et essais nouveaux ou à infléchir les programmes d'études et essais en cours ou prévus. L'accident survenu le 28 mars 1979 sur la deuxième tranche de la centrale nucléaire de Three Mile Island a conduit à procéder en France, dans le cadre des examens règlementaires de sûreté, à un premier examen des enseignements à en tirer pour ce qui concerne la sûreté des réacteurs à eau pressurisée et, parallèlement, à la lumière notamment de ce premier examen, à procéder à un infléchissement de certains programmes d'essais ; ainsi, la nécessité d'une meilleure connaissance des modes possibles de refroidissement d'un réacteur à eau pressurisée alors que le fluide caloporteur est diphasique a été particulièrement mise en évidence et un programme spécifique est en cours d'élaboration à ce sujet.

A ce stade, une remarque s'impose : si les examens règlementaires de sûreté interviennent essentiellement à certaines périodes de la vie d'une installation nucléaire définies le plus souvent dans le temps par les nécessités et les contingences industrielles et énergétiques, les études et essais en matière

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de sûreté ne font généralement évoluer que lentement l'état des connaissances, ce qui n'a pas d'inconvénient dès lors qu'est conservée à tout moment l'approche prudente décrite ci-dessus ; c'est ainsi que les études et essais particulièrement importants et nombreux poursuivis en France et dans le monde sur les conditions de dépressurisation des réacteurs à eau pressurisée et sur les conditions de renoyage du coeur d'un tel réacteur après dépressurisation ne font que lentement évoluer les analyses de sûreté relatives aux accidents de dépressurisation de ces réacteurs. Des problèmes de transposition de résultats d'essais, à première vue favorables ou défavorables, peuvent par ailleurs apparaître dès lors que l'on examine la sûreté d'installations nucléaires réelles. Ceci signifie que, si pour certains points, l'obtention de certains résultats d'études ou d'essais est considérée comme indispensable pour l'aboutissement favorable des examens règlementaires, il existe d'autres points qui seront examinés - et pourront l'être favorablement - sur la seule base des connaissances du moment, encore que, dans le but de mieux apprécier les marges réelles de sécurité, les autorités règlementaires puissent être amenées à lier la délivrance d'autorisations à l'obtention de certains résultats.
L'approfondissement des connaissances, but des études et essais de sûreté, peut en fait avoir pour premier fondement un objectif d'amélioration des installations, y compris sur le plan des performances, de la disponibilité et de la fiabilité. Les essais rappelés plus haut en cours dans la Chaufferie Avancée Prototype visent d'abord à permettre une meilleure utilisation des centrales nucléaires françaises, laquelle passe par une démonstration de la possibilité effective d'utiliser le combustible actuel dans un fonctionnement en téléréglage, démonstration où la sûreté a sa part. La définition d'un produit nouveau - tel qu'un combustible nouveau - suppose, dès lors qu'il s'agit d'un élément important pour la sûreté, une part d'études et/ou d'essais de sûreté ; bien entendu, ces études et essais sont effectués à l'initiative des constructeurs et exploitants mais les organismes de sûreté pourront y trouver un intérêt.

Le cadre dans lequel sont effectués les études et essais peut en fait être extrêmement variable et, en particulier, la part que prend l'Institut de Protection et de Sûreté Nucléaire dans les études et essais de sûreté menés en France dépend de nombreux facteurs. Ainsi, pour ce qui concerne les réacteurs à eau pressurisée, des éléments d'appréciation de la sûreté existent depuis longtemps compte tenu des nombreux réacteurs de ce type implantés dans le monde entier ; toutefois, des améliorations des connaissances apparaissent souhaitables et ceci justifie tout d'abord les études et essais relatifs aux accidents de dépressurisation qui font en France l'objet de programmes concertés entre les différents partenaires en vue notamment de mieux apprécier le comportement des éléments combustibles lors de tels accidents ; à cet égard,

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l'Institut de Protection et de Sûreté Nucléaire du Commissariat à l'Energie Atomique exploite à Cadarache un réacteur expérimental construit à cette fin, où un certain nombre d'essais seront effectués dans les prochaines années. Par ailleurs, l'Institut de Protection et de Sûreté Nucléaire a toujours accordé une grande importance au comportement de la première barrière opposée à la dissémination des produits de fission - c'est-à-dire les gaines des éléments combustibles - et des essais ont été entrepris depuis plusieurs années pour mieux connaître les mécanismes de relâchement des produits de fission dans le circuit primaire. Il faut d'ailleurs noter, sur un plan général, que le fait que de nombreux réacteurs d'un type donné soient exploités dans le monde entier ne dispense pas a priori de mener des essais et études de sûreté compte tenu de l'augmentation du nombre, de la taille des installations et surtout de l'évolution, en partie liée aux facteurs précédents, des idées en matière de sûreté ; à cet égard, on ne peut que constater qu'au cours des dix dernières années s'est affirmée avec de plus en plus de netteté l'exigence d'une sûreté bien clairement démontrée et de plus en plus quantifiée.

Dans le cas des réacteurs à neutrons rapides, les études et essais nécessaires à la démonstration de la sûreté de cette nouvelle filière sont en fait, pour ce qui concerne le Commissariat à l'Energie Atomique, partagées entre les unités chargées du développement et l'Institut de Protection et de Sûreté Nucléaire ; la règle générale adoptée consiste, pour préserver l'indépendance nécessaire de cet institut à l'égard des concepteurs, constructeurs et exploitants, à confier aux unités chargées du développement les études et essais plus directement liés à un projet d'installation tandis que l'Institut de Protection et de Sûreté Nucléaire réalise des études et essais plus fondamentaux indépendants en première approximation des détails de réalisation des installations mais susceptibles de faire largement évoluer la philosophie en matière de sûreté pour ces installations, ce qui rejoint le développement de la filière. C'est ainsi que l'Institut de Protection et de Sûreté Nucléaire exploite à Cadarache diverses installations visant à étudier le comportement des éléments combustibles lors d'accidents de refroidissement (réacteur Scarabée) ou de réactivité (réacteur Cabri) ; de la même façon, l'Institut de Protection et de Sûreté Nucléaire a réalisé de nombreux essais sur les feux de sodium et construit actuellement, toujours à Cadarache, l'installation Esmeralda en vue de mettre en oeuvre des quantités de sodium pouvant aller jusqu'à 70 tonnes, ce qui permettra d'une part de vérifier la validité des dispositions retenues pour la centrale nucléaire de Creys-Malville et d'autre part de procéder à des études complémentaires dans le cadre du développement de la filière des réacteurs à neutrons rapides refroidis au sodium. .../...

Dans tout ce qui précède, les exemples cités ont toujours été relatifs à des réacteurs nucléaires mais il va de soi que tout ce qui a été exposé s'applique en fait également aux autres installations du cycle du combustible et aux transports de substances radioactives, même si l'effort d'études et d'essais de sûreté est relativement moins important. On peut toutefois citer les essais effectués par l'Institut de Protection et de Sûreté Nucléaire à Valduc pour améliorer les connaissances en matière de criticité, par exemple les essais qui seront prochainement entrepris pour mieux apprécier l'influence combinée du gadolinium et du plutonium 240 en tant que poisons neutroniques ; ces essais sont directement liés à des questions soulevées lors des analyses de sûreté relatives au traitement des combustibles irradiés dans des réacteurs à neutrons rapides. On peut également citer les essais effectués à Cadarache sur le comportement de l'hexafluorure d'uranium dans l'air, essais là encore directement liés à des questions soulevées lors des examens règlementaires de sûreté de l'usine de séparation des isotopes de l'uranium du Tricastin. On peut enfin citer les essais relatifs à la migration souterraine du plutonium et des transuraniens dont l'intérêt est évident pour l'examen de la sûreté des stockages de déchets radioactifs.

S'il a beaucoup été question ci-dessus des études et essais de sûreté poursuivis en France, il faut noter qu'en fait les études et essais réalisés dans un pays donné doivent d'emblée être placés dans un cadre international. L'importance notamment financière de certains programmes d'essais justifie à l'évidence que les différents pays intéressés puissent y être associés selon des modalités à définir cas par cas ; ainsi, le réacteur Cabri dont il a été question ci-dessus est, à l'origine, une entreprise conjointe entre la France et la République Fédérale d'Allemagne, à laquelle sont maintenant associés le Japon, la Grande-Bretagne et les Etats-Unis d'Amérique ; l'Italie joue un rôle important dans la construction de l'installation Esmeralda. De nombreux accords d'échanges d'informations ont par ailleurs été signés par le Commissariat à l'Energie Atomique avec divers organismes étrangers pour échanger des résultats d'études ou essais ; à titre d'exemple, on peut citer l'accord conclu entre le Commissariat à l'Energie Atomique et l'United Kingdom Atomic Energy Authority sur les risques liés aux agressions de l'environnement et les accords plus larges conclus entre le Commissariat à l'Energie Atomique et les organismes de sûreté des Etats-Unis d'Amérique, de la République Fédérale d'Allemagne et du Japon.

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Il est clair par ailleurs que la définition des programmes d'études et essais de sûreté en France passe par la connaissance des études et essais entrepris dans le même domaine à l'étranger car d'une part il n'y a pas lieu très généralement d'effectuer les mêmes essais ou études dans différents pays, et d'autre part,dès lors que les résultats sont accessibles, une éventuelle duplication sera en tout état de cause plus riche d'enseignements si elle est accompagnée d'une confrontation des résultats. On rejoint ici un rôle fondamental des organisations internationales intéressées par le développement de l'énergie nucléaire qui consiste à créer les conditions d'un échange fructueux d'informations entre les différents pays.

CSNI SPECIALIST MEETING ON
REGULATORY REVIEW IN THE LICENSING PROCESS
(Madrid, 7-9 November 1979)

Relationship between Research and Regulatory Review in the Federal Republic of Germany

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Introduction

In the Federal Republic of Germany originally the nuclear research and the safety review of nuclear installations were under the competence and responsibility of one Minister, the Federal Minister for Education and Science (BMBW). The total Budget spent for nuclear safety research up to the year 1969 was only 5 million DM, which is equivalent to 2,9 million US $ (1979 exchange rate).

It was in 1970, when the plan of buildung the BASF-Nuclear Power Plant in Ludwigshafen caused a considerable increase in the yearly budget spent for nuclear safety research. The BASF-Project would have had a site with very high population and industrial installation density. In 1979 155 million DM equivalent 90 million US $ are spent for light water nuclear safety research.

Changing the less systematic choice of research projects before the Federal Minister for Education and Science in 1972 presented an overall Reactor Safety Research Program. In it, all problems concerning the safety of light water reactors were recognized and structured according to their importance. "Hypothetical addidents" were defined. Because of their extremely low probability, their risk was considered to be acceptable.

In order to distinguish research and review, in 1973 the responsibility for safety research was delegated to the Federal Minister of Research and Technology (BMFT). The responsibility for safety review - as far as it has to be done by the Federal administration - has been delegated to the Federal Minister of Interior (BMI) (see fig. 1). Caused by the German Atomic law, the responsibility for licensing and supervision of the licensee remains at different State Authorities.

1. The role of the Federal Minister of Research and Technology (BMFT) for reactor safety

BMFT plans and manages the German reactor safety research program, which is a part of the German energy research program.

The Federal Goverment accepts 24.00 MWe nuclear power capacity as a realistic - not necessarily satisfying - number in 1985. This is due to the recent energy crisis, the relative economical recession, the nuclear dialogue within the population and the political parties. Today 9.230 MWe (13 plants) are in operation. 14.780 MWe (again 13 plants) are in construction, two of them are being stopped by court trials.

It is not the goal of the safety research program to fullfil the licensing conditions of actual licensing processes. Its existence cannot be interpreted as a lack of information in todays licensing processes.

The goals and justifications of the nuclear safety research program are instead:

- The remaining residual risk for the population - caused by events, which are extremely seldom should be small in comparison to other, conventional risks accepted by the public, even with growing nuclear energy use.

- The radiation doses for the maintenance and repair personnel should be minimized by approved design and approved tools and techniques.

- According to the requirements of the Radiation Protection Ordinance, those fractions of radiation doses in the population caused ty the peaceful use of nuclear energy should be kept as low as possible - even beneath the dose limits defined in the Radiation Protection Ordinance.

  Following these goals the reliability of components and systems during operation, the reduction of accident probabilities and the knowledge of accident sequences and consequences are investigated. Methods for risk analyses are developed /1/.

  This is the basis for providing the licensing and supervision authorities as well as the independant experts with recent and future expertise, to support industry for organizing its safety technology and - last not least - for giving reliable information to the public and thereby achieving trust and understanding.

2. The role of the Federal Minister for Interior (BMI) for reactor safety

BMI supervises the licensing activities of the different state authorities for legal accuracy and practicability. That means, the states are submitted to BMI-supervision. BMI is entitled to give orders to the state authorities relative to nuclear safety and radiation protection /2/. The license itself is released by the different state authorities.

Equilizing safety requirements and safety qualities of installations in the whole Federal Republic is one of the main goal of BMI's activities.

2.1 The role of the Reactor Safety Commission (RSK), the Radiation Protection Commission (SSK), the Nuclear Engineering Committee (KTA) and the States Committee for Atomic Energy (LA)

a) The role of RSK and SSK

RSK and SSK are comparable to the ACRS in the US. They give independant advice to BMI.
The Reactor Safety Commission (RSK) treats safety questions in the nuclear licensing process. The Radiation Protection Commission (SSK) advises BMI for protection against the dangers of radiation.

The members of these commissions are independent and not subject to orders of any authority. Speaking within the RSK/SSK, they only represent their own expert's knowledge and skills. They do not represent the organizations they are usually working with. The results of these experts discussions are given as recommendations to BMI. BMI publishes them.

b) The role of the Nuclear Engineering Committee

KTA has been founded in 1972 by the Federal Minister for Education and Science (BMFT). 1974 the KTA was given under the control of BMI.

The main goal of KTA is to develop nuclear safety standards in those fields, which reached common and unique understanding by producers, constructors, users, independent experts and review authorities.

c) The role of the States Committee for Atomic Energy (LA)

The States Committee for Atomic Energy has been founded in order to establish a useful feedback of informations rising from the different licensing processes. BMI keeps the chairmanship of this Committee.

2.2 The Influence of advisory and regulatory committee on the safety research

Besides the broad German ligth water safety research program, there are safety related questions, risen by the work of RSK, SSK and to some extent by KTA and the LA. Further on, BMI is entitled to initiate investigations which may be integrated in the BMFT reactor safety research program (see fig. 2).

Initiatives for safety research projects are f.e. given by the Subcommittee on Safety Research of the RSK. BMI then decides about the future treatment of these suggestions, that means transmission to BMFT, beginning of research projects or else.

RSK f.e. has recommended 54 different themes for analyses studies, research and development projects in 1979, all concerning the safety of light water reactors. They have been classified following the scheme below:

A) Loads, design, supervision of electrical and mechanical components
B) Systems analyses, damage analyses, risk analyses
C) Advanced technical concepts
D) Analyses, studies and r+d-work concerning external events (mainly civil engineering)
E) Analyses, studies and r+d-work concerning transients
F) Analyses, Studies and r+d-work concerning loss of coolant accidents.
G) Analyses, studies and r+d-work concerning core melt
H) Influence of human behavior.

One example for the initiation of a research project by the RSK is the condition to find a certain minimum crack size during nondestructive testing of the reactor pressure vessel. During the time, when this requirement was formulated, no technical way to discover those little cracks was known. The relevant research program has become success meanwhile.

Out of 54 research recommendations of RSK in 1979 only 2 have been declared to be of direct importance for actual licensing activities (see fig. 3).

In general all proposals for research projects are examined in order to find out, whether the answers have been given elsewhere, whether the answer would be of help for reactor safety and whether a usefull final answer can be expected. It is important to notice, that BMFT has its own consulting committees for the light water reactor safety research program. Corresponding to the licensing character of this meeting, they are not shown or discussed in detail here /3/. The results of research activities financed by BMFT are all published.

3. Influence of research results on rules and regulations, guidelines and the design on nuclear power plants

In the same way as BMI and his advising committees influence the safety research and development, the results of these r+d efforts influence the activities and requirements of BMI and his committees.

The feedback of informations out of research into review is done in two ways:
a) the official information flow, characterized later on by the solid lines in fig. 4 and
b) the use of the professional background of the members of RSK, SSK, KTA and LA. Many of them come from those institutes, independent expert organisations and industries carrying out the research program. This is characterized by the dashed lines in fig. 4. Generating available highly qualified manpower for reactor safety purpose is therefore one of the most important relevances of the safety reserach program.

As far as necessary research results are integrated in updated formulations of the recommendations by RSK and SSK and of rules and regulations. All codes and standards are being revised as soon as technically necessary. F.e. RSK has revised its 1974 updating period of a basic set of guidelines certainly represents a high updating frequenzy.

Concluding it can be stated, that the basic structure of responsibilities in Germany separates safety research and safety review. There is an official exchange of informations between the authorities (BMFT, BMI, State Goverments) and their committees and Institutions (RSK, SSK, KTA, LA the independant experts, the central nuclear research centers etc.). Besides this, the many members of advising committees for the authorities come from organizations which carry out the reactor safety research program. Both effects make sure, that the licensing requirements recognize the state of the art as well as safety research recognizes the needs and future wishes of the safety review processes.

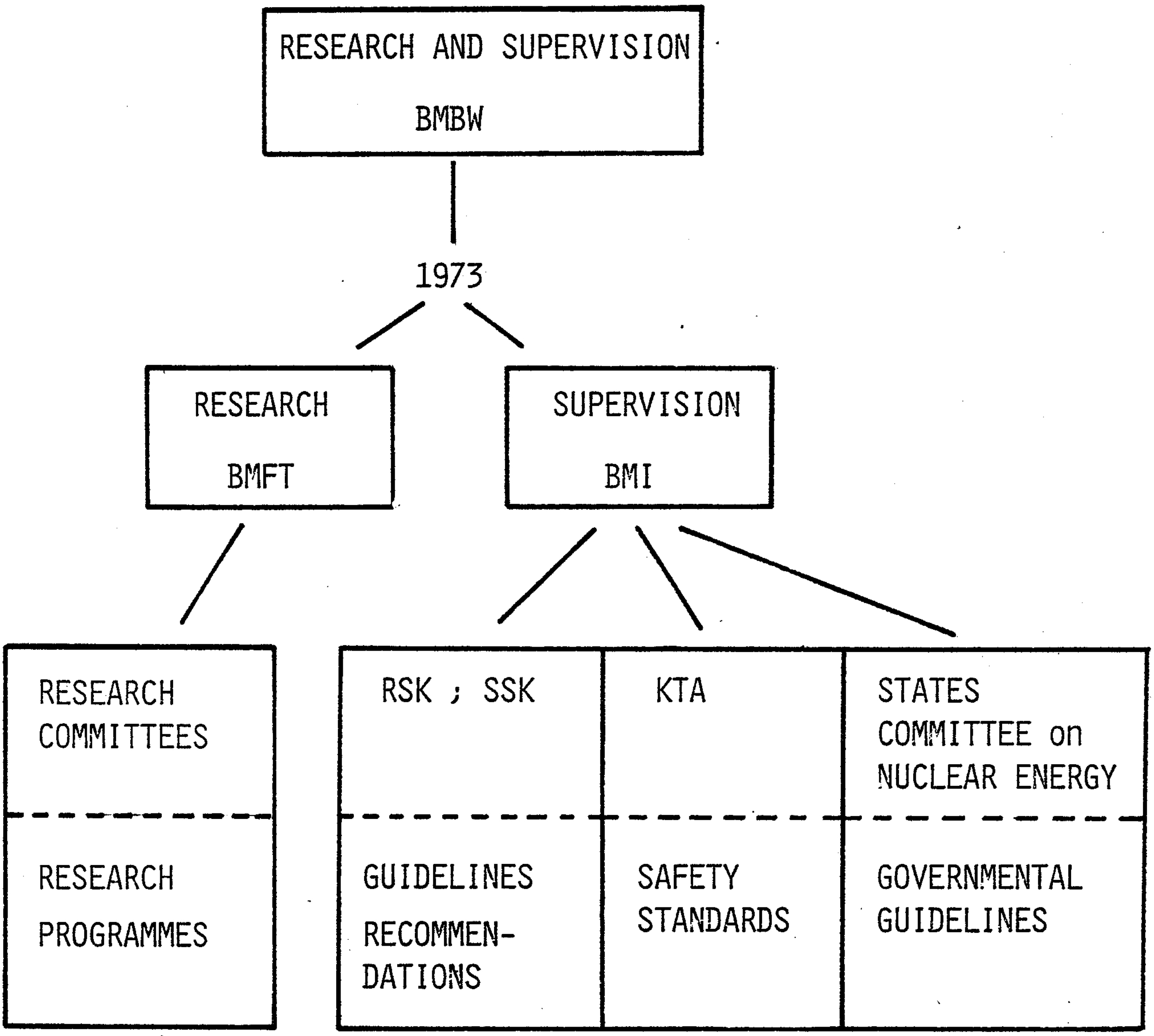
Literature

/1/ Assessment of Accident Risks from German Nuclear Power Plants
Dr. Heuser (FRG), CSNI Specialist Meeting, Nov. 79 Madrid

/2/ German Rules and Regulations with Special Reference to Application Documents
Dr. A. Kraut (FRG), CSNI Specialist Meeting, Nov. 79 Madrid

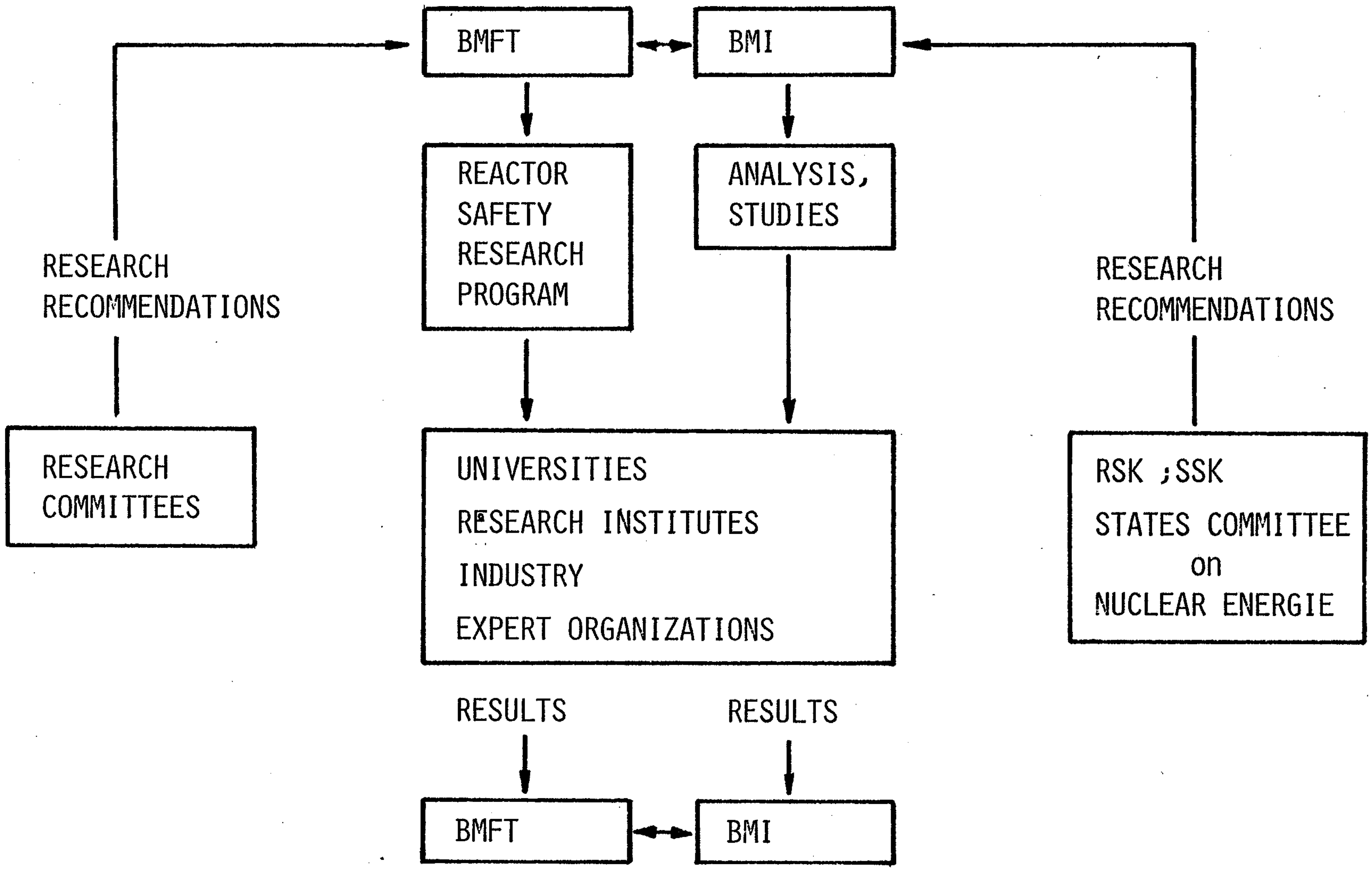
/3/ Beratungsplan 1979
des Bundesministers für Forschung und Technologie
BMFT, Bonn 1979

TASKS OF BMFT AND BMI



Picture 1

INFLUENCE of the ADVISORY BODY in the REACTOR SAFETY RESEARCH



Picture 2

RECOMMENDATIONS of the REACTOR SAFETY COMMISSION

for

REACTOR SAFETY RESEARCH PROGRAMMES

(JUNE 1979)

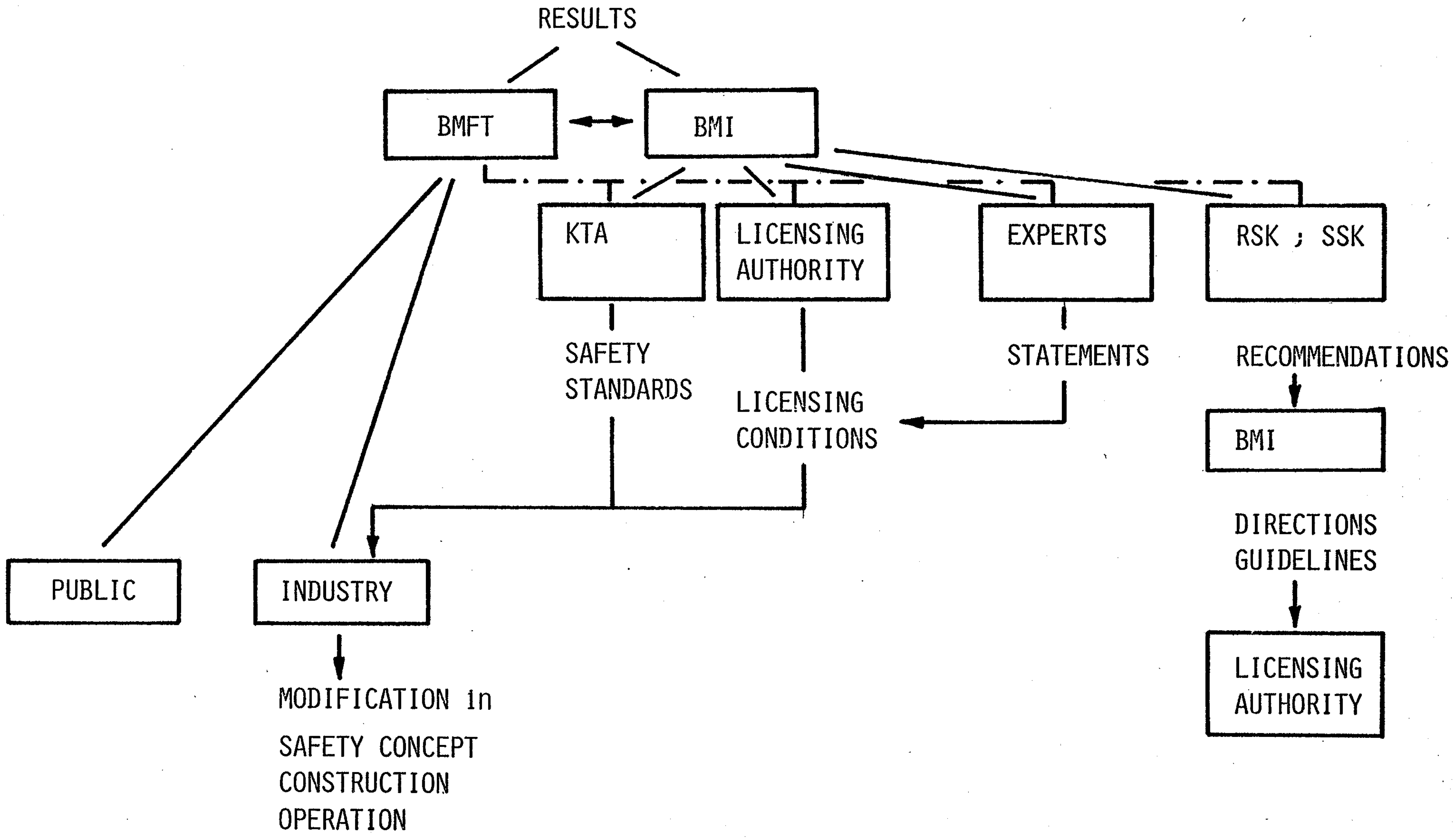
THE RESULTS OF THESE RESEARCH PROJECTS ARE RELEVANT FOR LICENSING AND SUPERVISION ACTIVITIES

- DOCUMENTATION AND EVALUATION OF THE RESULTS
  OF STRENGTH TESTS , BASIC INITIAL TESTS
  RECURRENT INSPECTIONS
  IN QUALITY ASSURENCE

- NONDESTRUCTIVE TESTING OF PREDAMAGED TEST SPECIMENS

Picture 3

INFLUENCE of the RESULTS of RS-RESEARCH on REGULATORY REVIEW



Picture 4

RESEARCH AND THE REGULATORY REVIEW

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To enable the regulatory review to be effectively undertaken by the regulatory body, there is a need for it to have ready access to information generated by research activities. Certain advantages have been seen to be gained by the regulatory body itself directly allocating and controlling some portion of these activities. The principal reasons for reaching this conclusion are summarised and a brief description of the Inspectorates directly sponsored programme outlined.

Afin que l'organisation régulatrice puisse se charger effectivement de la revue régulatrice, il faut que celui-là aît accès auprès des informations produites par les activités de recherche.

On a remarqué que certains avantages ont été gagnés par l'allocation et le contrôle directs de quelque partie de telles activités par l'organisation régulatrice.

Il y a un resumé ici-bas des raisons principaux pour lesquels on est parvenu à cette conclusion, ainsi qu'un court sommaire du programme de l'Inspectorat qui est directement fondé.

1. Introduction

Any Corporate Body wishing to construct or operate a commercial nuclear installation in the United Kingdom is required by law to obtain a licence from the Health and Safety Executive (HSE) through HM Nuclear Installations Inspectorate as a constituent part of this body. The way in which this process operates has been described in a paper previously presented in Session II and it can be seen from that paper that the approval of the regulatory body is required before the applicant can proceed beyond certain stages in the construction, commissioning or operation of any plant.

These approvals will only be granted by the HSE after it has carried out thorough reviews of all those features of the plant considered pertinent to maintaining safety, and relevant to the stage of the licensing process reached.

This regulatory review frequently reveals the need for additional knowledge and understanding which does not form part of the designer's safety case. Useful information can be gained from previous experience but it is often the case that the information required can only be obtained from further purpose built tests or basic research.

Much of this kind of work will be undertaken by the manufacturer or operator or will be carried out by other industry related research and development bodies such as the United Kingdom Atomic Energy Authority.

Examples of research sponsored by this latter body are given in the next and complementary paper.

There is however a need for the direct allocation and control of this type of work by the regulatory body itself and this latter approach has been seen in the United Kingdom to have certain advantages. The main reasons for reaching this conclusion, an outline of the advantages to be gained, together with a brief summary of the Inspectorate's own programme are given in this paper.

2. Need for Research Activity Input

It can be seen that, in a period of rapidly advancing technology and in an industry where a high level of technology is commonplace, it is essential that the regulatory body should have ready access to up-to-date information upon which to base its safety standards and ensure that adequate margins to safety are maintained. This is especially true for those features of a plant's construction or operation where there is little previous practical experience to fall back upon. But it can also be true for those more prosaic areas where performance has been taken somewhat for granted.

In particular the regulatory body requires to have:-

(i) information and understanding in good time ahead of the actual design work to help in establishing the principles and standards which will be applicable to the assessment of the construction and subsequent operation of new installations; and to set down the requirements for the safety case;

(ii) the earliest and fullest information concerning the safety and efficiency of existing types of installations so that any corrective actions or extension of controls or checks can be instituted quickly; and

(iii) independent sources of information upon which to base its own views.

Also the regulatory review will identify gaps in knowledge together with those areas of potential weakness or instability which may lead to hazardous situations. These gaps must be filled and the areas identified closely examined so that:-

(i) they can be shown ultimately not to be significant, either because of their intrinsic qualities or on account of the measures incorporated into the design; and hence be eliminated from further consideration;

or (ii) they can be shown to be sufficiently understood so that steps can be taken to bring about a modification to the construction or operational features of the plant so that the potential threat is eliminated or alternatively the consequences arising from a fault or failure can be contained.

hence the review can be completed.

To reach conclusions on these matters it is necessary to make judgements on complex and sometimes novel scientific and technical issues. Such judgements frequently reveal the need for further supporting information which is not currently available and which can only be acquired from research studies.

It can therefore be seen that one of the principal needs of the regulatory body is the promotion of such activities in the research field to enable the above duties to be carried out and a high standard of assessment to be maintained.

3. Advantages of direct control

The regulatory body could rely solely upon the results of research sponsored by the industry for carrying out its review but experience suggests that certain valuable benefits to be derived from a direct involvement in activities of this type would then be lost. These can be briefly outlined as follows:

Independence

In some cases judgements can only be made with confidence when supporting research is carried out on a truly independent basis. This can avoid potentially important decisions being based upon incomplete information; such a situation could arise using the results of industry based research which are frequently related to closely defined problems and are often influenced by commercial pressures. It is therefore considered essential that the regulatory body has the complete freedom of operation which can only be obtained by having direct control of both the technical and financial aspects of this type of work where it is thought necessary.

Fundamental Understanding

It has been found that direct involvement in research by the staff of the Inspectorate permits them to become better informed in the pertinent topics and allows them to appreciate with more clarity the issues involved.

This in turn permits the staff to take further part in discussions with the industry and to offer criticism or advice with enhanced confidence and credibility. In addition the fact that data can be made available from such work is a powerful factor in influencing decisions and persuading the industry towards a particular course of action.

Flexibility

Programmes structured by the Inspectorate's staff can be more closely tailored to fit their specific requirements on any particular issue or alternatively programmes can be loosely defined, if needs be, and allowed to develop over a broad and flexible basis. In this way again a more complete understanding of the problems can be gained by the staff due to the respectively more detailed or wider context in which they can be viewed.

Safety Awareness

Research programmes sponsored by the industry will have a tendency to be directed towards solving those problems associated with the need to successfully complete a project and demonstrate its efficacy. By orientating some programmes in the opposite sense the regulatory body can introduce a further element into the assessment process, and thus heighten the level of safety awareness in the industry. Even the knowledge that the Inspectorate is mounting an investigation into some particular facet has been found to create an interest within industry in its own right.

This awareness especially if re-inforced by the production of data obtained from investigation which shows that a question still has not been satisfactorily answered can act as a catalyst. Hence further and larger programmes of research into a topic can result from the industry itself taking a further interest in the topic and sponsoring its own programmes.

Special Investigations

Investigations which would normally be considered by the industry as not being "cost effective" can be mounted if considered appropriate. These types of investigations can cover certain unusual or extreme operational situations which may have been overlooked or considered by the industry as being of such a remote possibility as to require little or no detailed examination.

The regulatory body may not always share this latter view particularly where it is felt that the extrapolation of results from normal or upset conditions cannot be considered as reliable or feasible.

Also sensitivity studies to identify these parameters which will prove to be of most significance to the maintenance of safety can be carried out at the discretion of the Inspectorate using specific expertise only available from outside. Pilot studies can also be mounted if thought necessary to investigate the feasibility or credibility of techniques or methods available for solution of a problem. These techniques or methods may have either been submitted by the industry in support of a case or rejected by them as impracticable propositions.

Increased Resources

Work can be placed where a high level of expertise and competence in a specific topic exists and this enables information, facilities and resources not available within the HSE to be called upon. Familiarity with the factors of importance specific to the nuclear context may not exist in these groups but a learning process can be initiated and this in turn can create centres of excellence which can hence become more useful as further sources of independent advice and information.

4. Organisation and types of Research

Ideally it might be expected that the regulatory body should have its own research staff to conduct all the work thought necessary. However this can in many cases and certainly in the case of the Inspectorate be seen to be impracticable because of the wide range of expertise and facilities required, the limits placed on resources, and the under-utilisation of some forms of expertise which would result. It is also undesirable for the regulatory body itself to carry the major part of the responsibility for research since this responsibility must clearly lie with the manufacturers although some will be independent in origin. While an area of overlap between research activities is inevitable, the research sponsored by the regulatory body should be essentially exploratory or fundamental in nature by virtue of the above and because its resources are limited. The programmes should be aimed generally at showing a question requires answering rather than of attempting to provide an answer. Large programmes involving a heavy on-going commitment of resources, expensive rigs and long lead times are considered to be the responsibility of the industry itself in order to support its safety case.

Moreover the motivation for such research stems as much from the needs of the regulatory body's own staff to support their work of enforcement, setting of standards, giving guidance and drafting regulations, as from the needs of the industry itself.

The arrangement employed in the UK is to have a number of Project Officers, drawn from the staff of the Nuclear Inspectorate, each of whom is intimately involved in the safety assessment process in their specialist areas, appointed to direct and control the running of one or more projects. The identification of the projects and their contents is done by the staff and submitted for approval by management, after some preliminary consultation with possible contractors. After consideration by management of such factors as its importance, timeliness and cost, a project - if approved - is placed with the contractor where the expertise and resources are best thought available. This can include the Universities, independent research organisations, Government laboratories, or Government aided bodies like the United Kingdom Atomic Energy Authority. Independent consultants are also engaged in some cases and they can be involved intimately in the details of the project if thought appropriate. Projects are normally supported in the first instance for periods up to three years from budgets directly under the control of the Inspectorate.

5. Interagency Collaboration

A special area where more extensive commitments can be entered into is that where interagency and sometimes international collaboration can be shown to be appropriate and possible. Larger commitments on a shared basis can thus be entertained. There is a strong case for strengthening the existing interagency collaboration on safety research. This would require a recognition that the importance of safety, particularly as a matter of public re-assurance, is a question that merits co-ordination of effort, openness and sharing of results even if on an unequal basis. Duplication of effort can be avoided by this means and complementing of programmes can be arranged.

Examples of research where the kind of collaboration is possible are outlined in the following paper which illustrates some of the activities of the United Kingdom Atomic Energy Authority in this field. In such cases the Inspectorate may be represented on a number of research and development liaison committees to which it can introduce proposals and obtain access to results.

Further examples are to be found in the involvement by the Inspectorate's staff in European Economic Community and OECD sponsored research activities. The programme of work carried out at ISPRA and that of the HALDEN project in Norway are illustrations of this.

6. Outline of the Nuclear Installations Inspectorate Research Programme

The programme of research sponsored directly by the Nuclear Inspectorate forms a part of the overall research programme mounted by the Health and Safety Executive. This programme is of a very broad scope and covers all the areas which come within the regime of the United Kingdom's Health and Safety at Work Act. The programme is published annually prior to its implementation in the form of a handbook[1]. Each proposed project is listed with brief details of the reason why the project is being carried out, together with an outline of the proposed work content, and its expected timescale and cost. The listing includes those projects carried out 'in-house' i.e. within the Health and Safety Executive's laboratories, but does not include a set of fringe projects, closely allied to research but of a more practical nature which are placed under the heading of 'Support and Testing'.

The total budget for all the Research and Support and Testing activities of the HSE amounts to over £8.5m. The Inspectorate's direct share of these resources enables it to sponsor over 50 individual projects, a number of which will be on a shared basis with other organisations.

The Inspectorates programme covers a wide range of topics in its own right which range from the examination of the purely radiological aspects of safety to the study and analysis of complex engineered systems and components found in reactor plants. Both theoretical and experimental studies are sponsored; the theoretical studies being followed up with experimental work when considered appropriate and where resources permit. An important feature of this work is that although some projects are specifically nuclear in context many have a more general application owing to the wide range of advanced technologies applied to the total system. A similar cross fertilisation can occur from other projects carried out under the direction of the other divisions within the Health and Safety Executive.

The results obtained from this type of work are incorporated in the assessment or review process and disseminated by reports, published articles and presentations at International Conferences. Significant results from the past years activities are also precised in a further yearly publication by Her Majesty's Stationery Office[2].

Results are also directly shared in some instances with other agencies so that further work in the topic can proceed and a more comprehensive investigation result.

A listing of the projects mounted by the Inspectorate in the last year is provided in Appendix 1 for reference.

Conclusions

There is a need for the regulatory body to be directly involved in research activities if it is to effectively fulfil its statutory obligations relevant to safety. The ability to directly allocate and control a portion of these research activities can be seen to be of significant benefit to the regulatory body.

The type of research activities engaged in by the regulatory body should be confined to those which enable it to perform its statutory duties in an effective manner. Large scale programmes should be left for the industry itself to sponsor.

There is a powerful case for more interagency collaboration in this area in order that resources can be pooled and the information relevant to safety thereby increased.

References

1. Health and Safety Executive Research Programme - Research Planning Group, Health and Safety Executive, London.

2. Health and Safety Research 1976 et seq. Her Majesty's Stationery Office, London.

Appendix 1

Titles of projects directly sponsored by the Inspectorate in 1978 (excluding projects listed under Supported Testing).

Development of Forensic Techniques for Examining Corrosion and Oxidation Processes

The Effect of Combined Mechanical and Chemical Action on the Service Life of Nuclear Reactor Components

Effect of Chloride Contamination on Performance of Stainless Steel Components in AGR Environments

Avoiding HAZ Hydrogen Cracking in C-Mn Steels with Lean Alloy Additions

Significance of Arrested Short and Brittle Cracks in Fracture Toughness Testing of Weldments

The Role of Prediction in Establishing Siting Criteria

Thermal Explosions of Potential Significance to Nuclear Reactors

Fuel Pin Modelling Studies

Fast Reactor Whole-core Accidence Explosion Yield

Fatigue Crack Growth in CFR Materials

Liner/Concrete Interface Studies at High Temperatures

Core Configurations for a Low Sodium Void Reactivity Coefficient

Study of Energy Dissipation Structures

Non-destructive Examination of Fatigue Cracks in Austenitic Material under Sodium

Corrosion and Materials Properties of Steels in Sodium

Behaviour of Radioactive Isotopes in Sodium

Study of Fluid Dynamic Aspects of Energy Dissipation within the LMFBR Primary Vessel

High Temperature Sodium-concrete Interaction

Modelling of Reflood Heat Transfer

PWR LOCA/ECCS Refill Experiments

PWR Fuel Can Experiment (Flow Blockage)

Transient Dry-out and Flow Reversal

Upper Plenum Entrainment

Investigation of Computerised Methods of Event/Fault Tree Preparation

Transient Physics Studies

Pipe Whip Analysis and Energy Absorption

Probabilistic Fracture Mechanics

Radiological Consequences of Advanced Reactor Systems

Fuel Behaviour

Toughness of Stainless Steel at Elevated Temperatures

Ultrasonic Detection and Measurement of Defects in A533B Steel

Stability of Cracks in Tough Materials

Time Dependent Behaviour Study of Pre-stressed Concrete Pressure Vessel

SAFETY RESEARCH AND DEVELOPMENT IN RELATION

TO REGULATORY REVIEW IN THE UKAEA

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ABSTRACT

The basic functions of the Regulatory Review process are defined in principle, and the need for constructive interplay between regulation on the one hand and research and development on the other, in order to avoid the dangers of being over-restictive or too permissive, is established.

A number of practical examples from operating experience are quoted, which illustrate the chosen theme.

1. Introduction

The Regulatory Review process is concerned with ensuring:-

(a) that design and construction is in accordance with current standards and codes of practice where these exist and are applicable.

(b) that operating parameters are maintained at levels which provide a suitable margin against the safety limits derived from the design safety assessment.

(c) that these margins and limits are reviewed and revised when necessary in the light of operating experience and the results of technological progress, and in relation to improvements in understanding of the problems involved.

(d) that operational surveillance techniques are adequate and are periodically updated to take advantage of new developments.

(e) and that maintenance programmes are defined and observed so as to permit safe operation under economically efficient conditions throughout the life of the plant.

Clearly the regulatory function cannot be based on immutable precepts but rather must move with the times and take account of changes in acceptance criteria, of the availability of improved behavioural understanding, and of new theoretical studies, reliability, analytical and surveillance techniques. Indeed it is desirable that Regulatory Bodies play an active part in sponsoring and initiating R & D work in these fields. In the absence of such work the margins wich must be set in order to allow for uncertainties will continue to be either unnecessarily restrictive or unwisely permissive, neither of which extremes is consistent with safe operation at the highest possible economic level.

2. Illustrative Cases

Although UKAEA reactors are not subject to regulatory review by the Nuclear Installations Inspectorate, the objectives of such reviews are achieved through a network of Safety Committees and Working Parties on which the independant Safety and Reliability Directorate is represented. The function of these bodies is to advise management on the design and operational safety of the reactors and the need for special R & D support and to determine the content of Operating Rules and Safety Limits documents on which initial and continued authority to operate is based. Some examples of the interplay between R & D and regulation follow.

(a) Safety of Pressure Vessels

The earliest Stage I gas cooled graphite moderated Calder and Chapelcross reactor vessels were designed to BS1500 with generous margins in respect of choice of material and service conditions to provide reassurance against failure during their life by creep and stress rupture. Brittle fracture was recognised as a relevant factor and countered by guaranteeing to maintain the vessel always above its crack arrest temperature. Subsequent research and development in fracture mechanics identified the possibility of fast ductile fracture by the propagation of cracks greater than the critical crack length for the material, which is a function of fracture toughness. Where the initial pre-commissioning pressure test had been conducted at a pressure giving a safety factor of 1.6 - 2.0 times design pressure this could provide some reassurance against this failure mechanism. In the AEA the practise of conducting repeat proof tests at specified temperature and pressure conditions was developed in order to provide a period of immunity (the test interval) during which existing defects would not grow to critical length - or conversely to ensure that if the vessel might fail before the next test, it would do so during the test itself, with lesser safety consequences. Such considerations are of obvious relevance to the regulatory function and the periodic inspection/testing of reactor vessels is kept under continuous review by a Pressure Vessel Periodic Inspection Committee set up for each power reactor operated by UKAEA.

(b) Sodium-Water Reaction

The incompatibility of sodium and water and the violence of their interaction have always been of concern in respect of their separation by a single tube wall in modern heat exchanger design, partly because of the possibility of causing a pressure front in the primary circuit, and partly because of the possible extent of damage to surrounding tube bundles and the magnitude of reaction products.

In this context and in the absence of suitable research and development, the regulatory attitude could well be that where incompatible coolants are used provision must be made to prevent them mixing and to prevent harm to personnel and safety related structures in the event of them doing so.

The problem has been tackled in the NOAH and Small Water Leak Rigs at Dounreay, in which the extent of reaction resulting from injections of water into sodium through pin-hole defects for a time consistent with the detection and dumping system capability and response characteristics, has been studied and found to be not a serious objection to this design of heat exchanger.

(c) It can be a matter of safety principle on certain reactors that it should be possible for decay heat to be removed from a reactor to an adequate heat sink at any time in the life of the plant, irrespective of the availability of external resources. Credit can be claimed for natural circulation providing its effectiveness can be demonstrated in respect of adequacy, stability and reliability. Such a demonstration involves a combination of theoretical study and experimental tests in order to show that the mechanism is understood and that not only is the capacity for heat removed adequate, but that the transition form forced to natural convection is uninhibited by reversed differential heads etc. leading to an unacceptable stagnation period.

The first line of defence against loss of forced circulation in the PFR is via the three pony motor drives, there being a very low probability that the essential one from three will fail to clutch in as the main pumps coast down in speed. In spite of this the problem has been studied theoretically by the TRUDI and BUNTY Codes and some preliminary experiments have been done in a water rig which suggest that at very low flows the onset of boiling can stimulate the initiation of natural convection. The design of a sodium rig in which this work could be further developed for future reactors is well advanced. In addition a series of experiments in the reactor itself is in progress aimed at demonstrating the capacity for natural convection heat removal at successively higher powers and the smooth transition from forced to natural convection in the event of loss of electric supplies coupled with failure to engage at least one of the pony motor drives. It is intended to proceed with these tests so as ultimately to demonstrate natural convection capability and take up after a trip from full power, which would satisfy the quoted principle.

(d) Fuel Failure

The mechanism of fuel pin failure in fast reactors and the possibility of blockage by the resultant debris, restricted coolant flow leading to overheating, and escalation of the failure to involve the remainder of the sub-assembly has always been a matter of concern in safety studies. From the regulatory point of view it might be suggested as a principle that the protection system must detect the onset of pin failure and shut the reactor down with adequate diversity/redundancy in detection and corrective action, speed of response etc. The important parameter here is the speed of response which is required in relation to the rate of development or deterioration of the initial defect, its effect on coolant flow through a partial blockage, the effect of local boiling etc.

Such information can only be provided by a combination of rig and reactor tests as part of the overall Safety R & D Programme to determine the extent to which a real problem exists and to guide the protection standards to be set.

In this context the opportunity was taken in the closing stages of DFR operation, to conduct a series of special tests involving the deliberate reduction of flow in experimental sub assemblies so as to provoke sodium boiling, to observe the stability or otherwise in that regime, and to discover whether rapid deterioration leading to escalation of the fault could occur. The results of this programme demonstrated the ability of LMFBR fuel bundles to tolerate severe localized faults without rapid deterioration into a possibly undetectable and propagating incident. Support has been lent to this conclusion by additional and exceedingly severe tests on a fuel pin bundle in another reactor which was operated at full power with 35% flow area blockage plus some secondary blockage present without further observable deterioration. This work has contributed to the view that the sub-assembly incident is of less concern than previously imagined in relation to the need for very high speed detection and shutdown to prevent escalation, and there is thus no need for such provisions to be made as a matter of principle.

(e) Containment

Regulatory principles will require that the containment building will be capable of protecting the public from the escape of any radioactive materials within it with high probability of success and of withstanding the effect of faults, account being taken inter alia of any impulse loading, missiles, explosives etc. arising from such faults. These aspects are notoriously difficult to demonstrate adequately by theoretical calculations. In order to supplement the latter, research and development is in progress on two aspects. Experiments are being conducted in a Containment Modelling Rig in which explosive charges are fired inside modelled containment structures, suitably instrumented to provide information on the pressure and strains produced; and a Missile Launcher is available which uses compressed air to fire projectiles at containment specimens. The outcome of such experiments will go a long way towards providing evidence that the containment function will be fulfilled in the context of regulatory requirements and public safety.

Conclusion

The foregoing was not intended to be a survey of R & D work in progress in the UK but rather to illustrate the type of work required and the way in which it needs to be used to support the safety case, particularly in areas outwith normal codes and standards in order to meet the spirit and letter of regulatory requirements.

CSNI SPECIALIST MEETING ON
REGULATORY REVIEW IN THE LICENSING PROCESS
(Madrid, 7-9 November 1979)

CURRENT ROLE OF THE USNRC SAFETY RESEARCH PROGRAM IN SUPPORT OF THE REGULATORY PROCESS

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I am pleased to be here to speak to you today on the current role of the USNRC's safety research program. In discussing some aspects of this role in the wake of the TMI accident, I would first like to cover some historical perspective on the development of our program, its relationship to the NRC mission, an overview of the program activities and some recent research results, and finally the impact of the TMI accident in clarifying needs for expedited and new research activities, including the need for a greatly enhanced use of probabilistic analysis techniques to improve the coherence of our regulatory process.

Perspective

Some historical perspective on the establishment of the Office of Nuclear Regulatory Research (RES) as part of the USNRC is provided in Figure 1. The severe criticism of nuclear power in the U.S. that began in the late 1960s, had developed by 1972 into serious discussions between the then existing USAEC and the scientific community and public over the safety and environmental impacts of nuclear power, with specific attention given in public hearings to the question of ECCS acceptance criteria. The response of the AEC provided for an overall strengthening of the regulatory program, the creation of a new and separate Division of Reactor Safety Research, the preparation of environmental impact statements as required by law, and, in 1972, the initiation of the Reactor Safety Study under Professor Rasmussen of MIT, directed to making a quantitative analysis of the risks to the public from potential accidents in U.S. commercial nuclear power plants.

The Reorganization Act of 1974 that created the NRC as an independent agency of the government recognized its need to have an adequate, independent research capability by establishing the Office of Nuclear Regulatory Research (RES) as part of the NRC. It transferred the reactor safety research and risk assessment functions from the AEC to the NRC, and recognized NRC's need to conduct research in the safeguards, fuel cycle, and environmental areas.

As originally delineated, NRC's research role was limited to confirmatory research only. This requirement kept NRC's research program in a primarily reactive mode, with little initiative to conduct research in areas that could lead to the development of improved reactor safety systems. As discussed below, this role was subsequently modified by Congress in 1978 to allow NRC to explore and evaluate new concepts for improving the safety of nuclear power plants.

<u>Relationship of Research Program to NRC Mission</u>

As shown in Figure 2, the work performed in our research program is in response to stated and perceived licensing and other regulatory needs. The program covers all aspects of the nuclear fuel cycle except for uranium mining, which is regulated by the states. As indicated in Figure 3, our program responds to needs generated by the major NRC program offices, the various regulatory boards, the Advisory Committee on Reactor Safeguards, and the comments of the scientific community and interested public.

The procedure used within NRC to ensure the relevance of the research work to our long- and short-term regulatory requirements is an integral part of the annual budget process of the NRC. For the safety research program, this involves the development of the technical content of the program, the assignment of program priorities, a series of iterative review actions by OMB and the Congress, and ultimately, the allocation of funds to RES.

In arriving at the approved program of activities and in implementing them, the procedure we use involves the following coordination among the five major NRC program offices (i.e., Reactor Regulation, Material Safety and Safeguards, Inspection and Enforcement, Standards Development and Research):

(1) First, there is the identification of a research need or objective by one of the major program offices, this identification being formulated as a "user research request" and formally transmitted from the requesting office to RES. This user request must clearly delineate the need and usefulness of the proposed research and provide sufficient information to permit specifying an appropriate research project responsive to the indicated need. In many cases, the user requests sent to RES are based on needs perceived by the relevant competent technical people in RES.

(2) The second step calls for RES to convert the user research request into a specific definition of work, with the scope, level of effort and timing appropriate to the nature of the request. This definition of work receives a formal endorsement from the requesting user office before contract action is taken, in order to signify the concurrence of the requesting office that the proposed RES plan meets the defined need.

(3) During the research implementation phase, RES undertakes continued coordination and information exchange with the sponsoring office through various means, such as routine staff interactions, reviews of research activities, issuance of Research Information Letters, contractor publications and papers, and, in particular, by meetings of the research review groups established for each of the program activities. The NRC has created about 70 research review groups to cover the various disciplines included in our research programs. These research review groups, which include some of the best available researchers in each field, play an important role not only for internal coordination purposes but also as a mechanism to assist RES in guiding its research programs and in providing for peer review.

While this user request approach has worked reasonably well in helping to ensure the relevance of our research program to agency needs, experience has made visible two severe difficulties that arise from its use. The first involves the procedural requirements used for endorsing research projects, which in the context of the Commission's budget process, can act to introduce an undesirable degree of inflexibility and delayed responsiveness in accommodating the research program to changing requirements. The second is the difficulty of achieving an appropriate balance between the generally narrower, shorter term interests of the licensing offices and the broader, more fundamental scientific views of the research office directed to ensuring the agency's capability to handle broad ranges of problems.

In this connection, it seems clear that a closer conjunction is needed between the research and licensing thought processes. In the past, licensing safety decisions on light water reactors have been made conservatively as a matter of prudence and to account for gaps in knowledge about the various physical phenomena related to accidents. This has led to some confusion between those who license reactors and those who perform the research needed to support the licensing of reactors. Researchers must necessarily think about the relevant accident phenomena in realistic terms because they have to plan real experiments and develop models that describe that reality well enough so that the results of experiments can be pre-predicted and be confidently extrapolated to cover the behavior of large reactors. On the other hand, licensers tend to think too heavily about regulatory models and criteria which are necessarily designed to be conservative. In fact these conservative thought processes caused us some significant difficulties during the Three Mile Island accident.

The principal thrust of the light water research conducted over the last few years has been to define with greater precision the safety margins provided in reactor designs by the established conservative licensing approach. Indeed, this objective is common to the large-scale international research programs associated with ECCS. The adoption of this research approach in conjunction with the established licensing approach has been generally agreed to by the scientific community in the U.S. on the basis that, while all the realistic data and models are not yet available, there are research programs in place directed toward that end. At the same time, one must take note of the growing body of opinion among some people, primarily outside the technical community, to the effect that technically complete answers to safety questions must be on hand before any licensing decisions are made.

The essence of the message we seem to be getting is that we now have to have more scientific information about the safety of nuclear power available at the time decisions are made than we have had in the past. Achieving this objective is, of course, quite difficult. It clearly has to involve a much closer cooperation and detailed technical thinking by those involved in licensing and those involved in research. In general, licensing people are enmeshed in the day-to-day problems involved in issuing construction permits, operating licenses, changes to licenses, and operating problems in reactors. This makes it difficult for them to have the time to think about and define the long-term needs affecting future potential problems. Research takes some time to do and large research programs involving construction of facilities can take years. Thus, the timescales of the two thought processes do not tend to be compatible. On the other side, many of the research people have quite narrow parochial viewpoints that do not permit them to fully understand the details and needs of our regulatory activities.

Probably the best way to correct these problems is to give our research program more flexibility. The need for some clarification and expansion of our user requests policy has now been recognized within the NRC and changes are being planned. The planned revision of the policy will include, as one element of the policy, the current user requirement procedure described above for most of our research projects. This will also include endorsements that cover the individual projects over their lifetimes, as opposed to requiring annual reendorsement. It will also include programmatic as opposed to just project endorsements. This would enable the NRC user office to endorse large RES programs (LOFT, PBF) made up of multiple projects, where the user offices have no direct use for a specific result but agrees that the agency needs the program. Finally, it will also include a proposal to allow RES itself to endorse up to 15-30 percent of the research program in order to permit RES adequate flexibility for better ensuring the adequacy of fulfillment of our longer range needs and to allow for rapid response to new problems.

Overview of the NRC Research Program

In the implementation of its research charter, NRC has considered three types of research, distinguished by the different goals embodied in each: (a) developmental research, (b) confirmatory research, and (c) research for improved safety.

Developmental research is research conducted to evaluate the safety of materials, processes, and equipment likely to be proposed by an applicant for an NRC license. It includes research performed in the process of developing and designing a proposed facility, as well as any research needed to provide information in support of a safety assessment. This type of research is not performed by the NRC, but is performed by industry and the U.S. Department of Energy.

Confirmatory research, which largely comprises the current NRC program, is directed to providing NRC with an independent, confirmatory capability for evaluating licensing applications and regulating the use of nuclear power and materials, as well as providing a basis for regulatory requirements or policy. In many areas, the results of confirmatory research also provide an important basis for improved reactor safety. It is carried out by NRC independently of the nuclear industry. As shown in Figure 4, the confirmatory program is aimed at providing objectively verified safety data and analytical methods required for NRC's regulatory activities. Since licensing activities are based on conservative approaches, these data and analytical models permit more realistic estimates of the performance of safety features than those provided by the regulatory evaluation models, and thereby permit a measure of the safety margins implicit in licensed facilities and operations.

As indicated in Figure 5, the major activities of our confirmatory research relate to reactor safety, safeguards, the fuel cycle safety and environmental aspects, and risk assessment.

As an example of the basic logic involved in the development of verified analytical methods for regulatory use, Figure 6 shows in flow chart form the work being performed in the area of ECCS research. The separate effects tests are designed to provide data to permit the modeling of individual LOCA phenomena of interest. The integral tests in Semiscale and LOFT provide data on the behavior of the overall plant system during a LOCA, while the PBF facility is used to obtain test data on fuel behavior under LOCA conditions. These experiments are providing data for the development of physical models and for overall tests of ECCS codes. Since not all the experimental data may be available for initial closure of work leading to fully verified codes, and since it is possible that additional work may be required, a small effort for a second closure is shown.

Research for improved safety is an additional segment of the NRC program. It is responsive to NRC's amended research charter in the Fiscal Year 1978 Budget Authorization Act of Congress, requiring NRC to develop a long-range plan for the development of advanced concepts, systems and processes believed to have potential for improving the safety of nuclear power. The purpose of the plan is to investigate the feasibility, benefits, and costs involved in the application of such concepts.

As reported in NUREG-0438, NRC's work in this area has involved assessment of a large number of proposed research projects examined against a set of judgmental criteria consisting of the breadth of technical support, the potential for reducing reactor accident risks, the generic applicability of potential improvements and the estimated cost of implementation. A recommendation was made for initial research on the following topics listed in order of priority:

- Improved in-plant accident response, to reduce the risk of human error by reducing test and maintenance errors and by helping operators make correct decisions during accidents.

- Alternate containment concepts, especially vented containments, to mitigate the consequences of postulated core meltdown accidents.

- Alternate decay heat removal systems, especially add-on bunkered systems, to reduce the probability of core meltdowns by increasing the reliability of systems designed to remove residual heat from the shutdown reactor core.

- Alternate emergency core cooling concepts, to develop simpler and more clearly demonstrable systems to prevent fuel overheating in the event of pipe rupture.

- Advanced seismic designs, to reduce the vulnerability of plants to earthquakes by decoupling or strengthening components against seismic forces.

Work on such projects would be directed to producing safety system design and performance requirements and value/impact analyses associated with their implementation in plant. Actual implementation would require the establishment of regulatory criteria or rules by NRC, and an NRC review of detailed industry proposals to assure compliance with the regulatory requirements.

Use of Research Results in Support of NRC Regulatory Activities

As indicated above, upon completion of a substantial, coherent and reasonably complete piece of research, RES prepares a Research Information Letter (RIL) to the requesting NRC program office(s). A RIL may cover material developed from more than one research project. The user office reviews the information contained in the RIL and considers its utilization for regulatory activities. The appendix to this paper contains a list of the approximately 60 RILs issued to date.

Our research results are being used in a number of ways, including: as a technical basis for regulatory rules, guides and standards, for establishing technical specifications, for evaluating vendor licensing calculations and design features, for site evaluations, evaluation of generic safety issues, risk assessments, improvement of safety, etc.

In the short time available to me here, I would like to discuss only several of the important research results obtained, and how these have been utilized in support of our regulatory activities:

(a) Decay Heat Rate (RIL No. 8): The NRC acceptance criteria for LWR ECC systems require that the fission product decay heat rate assumed for a postulated LOCA be 20% larger than the standard approved by the American Nuclear Society (ANS) in 1971, which had been judged to have an uncertainty in decay heat value approaching 15%, especially at cooling times of less than 100 seconds. More recent experimental and analytical work has demonstrated the conservativeness of the ANS standard during the first seconds after shutdown and shown that the uncertainty is less than 5% at short cooling times and decreases as the cooling time increases. On the basis of these results a new standard, approved in August 1979 by the ANS Board of Standards Review will be evaluated by NRC in considering amendments to its Appendix K acceptance criteria.

(b) Zircaloy Oxidation (RIL No. 9): In the analysis of accidents to assess ECCS performance, the oxidation of zircaloy in steam is an important phenomenon because of the hydrogen and heat generated by the exothermic reaction, because the oxidation of the cladding reduces the wall thickness capable of carrying tensile stresses, and because of the possibility for distortion (ballooning) of the cladding.

Research results from ORNL on the oxidation rate of zircaloy show that the rate at high temperature (about 2200°F) is only about 60% of the value obtained from the Baker-Just equation. Results from several separate investigations, including data from BMFT- and JAERI-sponsored tests, have yielded a new rate equation considered more realistic than the Baker-Just equation currently used in licensing evaluation calculations. As a result, calculated peak cladding temperatures obtained for a postulated LOCA are estimated to be about 100°F lower with the new equation than with the older more conservative one. Furthermore, experiments have shown that with the lower oxidation rates, an autocatalytic metal-water reaction should not take place even at temperatures as high as 2300°F.

Since zircaloy becomes embrittled as it oxidizes, it may not withstand the thermal shock due to the quenching action of the injected ECC water. The outer layer of the zircaloy clad will become heavily oxidized and greatly embrittle the clad to some depth depending on the amount of zircaloy-water reaction that takes place. However, the remaining unoxidized beta phase zircaloy is expected to remain sufficiently ductile to withstand the thermal shock forces. Oxygen diffusing into the beta zircaloy ahead of the heavily oxidized layer reduces this ductility. The latest experimental results show that the rate of diffusion of oxygen into beta zircaloy is only half of the value used in licensing evaluation calculations. Thus, there appears to be significant conservatism in this area in current calculations.

The mechanical behavior of zircaloy cladding has shown less ballooning than was observed in previous tests, which were conducted in an inert as opposed to a steam atmosphere, with unrealistic internal gas volumes and axial constraint conditions. These test results mean there will be less likelihood of the clad ballooning so as to block the flow of cooling water in a LOCA.

(c) Reactor Pressure Vessel Integrity (RILs Nos. 1, 3, 5, 10): Analytical methods for predicting the stress-temperature flaw size relationships that could cause failure of thick-walled pressure vessels have been under extensive development over the past few decades. The fracture mechanics methodology used for such predictions has been developed for the linear elastic range of material behavior and, more recently, for the elastic-plastic range which corresponds to higher stress levels. The landmark validation of this methodology has come from the tests to failure of eight intermediate-scale, thick-walled pressure vessels, performed as part of the HSST Program. The wall thicknesses used (6 inches) were nearly the same as used in actual reactor pressure vessels. In all cases, the failure condition of each of the deliberately flawed vessels was accurately predicted by the linear elastic and/or elastic-plastic fracture mechanics methods. As these techniques form the basis for calculations used to set operating limits for reactor pressure vessels related to allowable pressures below the nil-ductility transition temperature, a computer program (OCTAVIA) was developed to enable a detailed examination of this area.

The OCTAVIA code first calculates the pressure-temperature-flaw size relationships for failure of reactor pressure vessels, given the input characteristics of neutron fluence level, vessel geometry, strength level of the unirradiated steel, copper and phosphorus content of the steel and initial RT_{NDT} (reference nil-ductility transition temperature). It then calculates a best-estimate failure probability as a function of vessel pressure based on (1) the probability of occurrence of over-pressurization transients derived from the actual number of such events that have occurred during the startup and shutdown of PWR plants in service, (2) the statistical distribution of vessel temperatures and pressures that have occurred in these transients, and (3) the estimated efficiency of flaw detection in inspection as a function of flaw sizes that might exist in vessel walls.

The development of this code is of particular significance in that it has brought together in an important way the application of probabilistic techniques to the recently completed results of physical research. An examination of the Surry 1 reactor vessel using the OCTAVIA code predicts a failure probability, from the type of overpressurization events that have occurred in PWRs, that lies within the range of vessel failure probability (10^{-8}-10^{-6}) predicted in the Reactor Safety Study. The work also shows that failure probabilities will increase significantly with vessel irradiation and that actions that the regulatory staff has underway should be able to reduce end-of-life failure probabilities significantly.

The HSST test results have also given us an answer to the question posed by the ACRS as to a possible difference in failure modes between a vessel failing under hydraulic loading and one failing under a sustained pneumatic loading, as in a vessel with a hot steam-water mixture. Tests of two vessels with identical flaws, one tested with cold water and one with N_2 gas to simulate a steam-water mixture, failed at identical pressures and in identical ways.

(d) FRAP Fuel Behavior Computer Codes (RIL Nos. 25, 29, 59): The assessment of fuel rod integrity necessitates fuel codes capable of analyzing the combined thermal, mechanical and internal gas behavior of fuel rods under normal operational and accident conditions. The aspect of fuel rod thermal behavior involves the correct modeling of the power distribution in the fuel pellet, the thermal conductivity of the fuel and cladding, and the transfer of heat across the pellet-cladding gap and from the cladding surface. Fuel rod mechanical behavior involves the correct modeling of the pellet-cladding mechanical interaction, the occurrence of any creep, ballooning and failure effects in the cladding, and the effects of thermal expansion, swelling, densification and creep in the fuel pellet. The correct modeling of the internal gas, which determines the pressure loading on the cladding and the heat transfer that occurs across the gap, includes the important factors of axial gas flow, fission gas release, plenum gas temperature, and voids and void temperature.

Two basic fuel behavior codes developed under our research program are FRAP-5, for steady state operation, and FRAP-T, for analysis of transient accident conditions. FRAP-S with either a BE or EM module, can be used as an operation analysis tool or for calculating the burnup-dependent initial conditions required as input for the FRAP-T accident code.

The FRAPCON-1 code (RIL to be issued) is a steady state fuel behavior code, representing a merger of the two base codes FRAP-S and GAPCON-THERMAL, optimized for ease of use and running time, and modularized for easy interchange of EM and BE models. It can be used to determine the input condition for the applications of the BE FRAP transient code or used as a licensing tool with appropriate EM models for regulatory auditing. Versions of these codes up through FRAP-S3, FRAPCON-1 and FRAP-T4 are currently available for use from the National Energy Software Center at ANL.

(e) WRAP-EM Computer Program (RIL to be issued): The Water Reactor Analysis Program (WRAP) is a package of evaluation model codes, assembled to provide greater convenience of application of those codes which are used in the licensing review process. Figure 7 illustrates schematically, the automatic linkage of various codes for the case of the PWR LOCA analysis. Here, the initial conditions on stored energy in the fuel obtained with the GAPCON code are automatically provided as input to the blowdown code RELAP-4/MOD 5, which is linked through a refill module to the reflood code RELAP 4/Flood and to the hot channel code FRAP-T-EM. A version of WRAP for BWR-LOCA audit analysis is also available.

(f) FRANTIC Computer Code (RIL No. 18): This code has been developed to calculate the detailed unavailability of safety systems. FRANTIC calculates not only the average unavailability but also the time-dependent instantaneous unavailability of a system. Even though a system has a low probability of being unavailable when averaged over a year, it could have a high instantaneous unavailability at a particular time. The time-dependent instantaneous unavailability thus gives a detailed picture of the readiness of the system to respond should an accident occur at any given time.

The FRANTIC code includes detailed effects of periodic system testing. The testing characteristics considered by FRANTIC include the test interval, the test duration, the repair time or allowed downtime, the test override capability, the test efficiency, and the probabilities of man-caused failures associated with the test. As an example, FRANTIC was used to evaluate the auxiliary feedwater system analyzed in the Reactor Safety Study; the results of both analyses were in agreement.

A sensitivity study of the reliability of the auxiliary feedwater system was also performed, evaluating the effect of various testing schemes on feedwater unavailability. By optimizing the testing schemes, it was found that system availability could be improved somewhat. More to the point, the use of optimum testing for the Reactor Safety Study accident sequence involving the auxiliary feedwater system showed that the average probability of this sequence could be decreased by as much as a factor of 20 and the peak instantaneous unavailability could be decreased by as much as a factor of 40. The optimum involved adjusting the scheduling of tests but not the frequency of testing. In fact, from this simple example, we gained the general insight that large benefits may be obtained by staggering tests not only within the same system but also across different systems in the same accident sequence. FRANTIC will be used as an aid in establishing a better basis for limiting conditions of operation and for testing requirements in technical specifications for nuclear power plant licenses.

Impact of TMI on Research Needs

Design basis accidents (DBAs), such as the large break LOCA, have been studied extensively in the licensing process to ensure that plant safety equipment (ECCS) have adequate safety margins to prevent fuel damage in the event of a DBA. The TMI accident was one which, progressed from a loss-of-main-feedwater transient to a small LOCA and thence to core uncovery and subsequent fuel damage. The occurrence of TMI has therefore emphasized the urgent need for additional safety research on accidents which can lead to extensive core damage. This category of accidents is indicated schematically in Figure 8 in the area between the DBA and core melt accident regions. Core melt accidents were examined in the Reactor Safety Study (RSS) and are currently being studied experimentally and analytically to better define various processes, including fuel melt debris bed coolability limits and extended dryout behavior; the physical phenomena involved in the interactions between molten fuel and structural materials (concrete, steel, refractory and sacrificial); the explosive interaction between molten fuel and coolant; the release and transport of radionuclides from the reactor fuel, and the consequences to the public.

The studies of core melt accidents in the RSS involved assigning failure probabilities to various safety systems, the failure of which would lead to core melting, affect containment integrity and the amount of radioactive material released to the environment. The risk assessment studies addressed the probability and consequences of core melt accidents.

The RSS did not extensively examine accidents involving core damage without significant fuel melting, because such accidents were not thought to have large public health consequences, on the basis that lack of a molten core would not cause a threat to containment integrity. The indicated area between the DBAs and core melt accidents has accordingly tended to receive insufficient attention in both the licensing process and in the research programs of NRC. Such accidents, similar to TMI, can occur as a result of partial failure and intermittent operation of various systems and may lead to extensive core damage, with fuel melting. It follows that the TMI accident sequence was not a unique one and that other similar accidents starting from a variety of operational transients and leading to extensive core damage can be postulated.

Accordingly, it is clear that small LOCA, transient events and enhanced operator capability are areas that need additional research resources. In particular, better computer codes are needed (1) to enhance our understanding of small LOCAs and transients, (2) to allow multitudinous studies to be made of these types of events and the many variations that can occur in them, and (3) to predict with greater precision than now obtainable the behavior of plants in response to such events. The development and checking of these codes will require experiments in such facilities as LOFT and Semiscale (for PWRs) and TLTA (for BWRs) to provide insights to develop the physical models in the codes and to check their range of applicability. The availability of these same codes will allow studies to be made toward enhancing operator capability. Studies will be made of simulator requirements to enhance their capabilities for training plant operators, analyses of the instrumentation needed by operators to understand and react properly to the full spectrum of potential reactor accidents, and studies of the control room display and diagnostic equipment needed to assist the plant operators in effecting proper responses and ensuring that limiting conditions of operation are met. In this connection, there is the need to establish an adequate data link between the plant and the regulatory and other outside organizations capable of providing assistance and advice to the reactor operator in the event of an emergency. In addition, these same codes will allow us to analyze the startup transient tests already performed on operating reactors and will give NRC the understanding and the basis for specifying additional startup tests that may be needed on operating plants. At the same time, risk assessment tasks to construct event trees are needed to define accident sequences covering severe core damage which the codes must calculate and to guide the research tasks needed to assess the potential impacts of human errors on the course of these types of accidents. In parallel with these studies it is necessary to investigate potential means of improving plant design features such as improved decay heat removal and ECC systems, vented containment concepts, etc.

Another area of great interest is the need to better understand the response of plants to accidents of the kind that occurred at TMI. It is clear that we need a better understanding of primary coolant chemistry after severe fuel damage, hydrogen evolution and behavior in the primary coolant system and in the containment, behavior of important plant components under long term, severe accident environments, equipment qualification and testing requirements and structural analysis of important plant components and safety features under accident conditions.

Finally, it is important to preserve the data on the amount and dispersion of fission products throughout the plant and to examine the TMI fuel to assess the type and extent of damage to the core. In parallel, it will be necessary to examine safety related equipment in the plant to assess the extent of damage and to establish criteria for safety requalification of the plant.

On this basis, NRC is planning research in the following areas:

(a) Transient and Small LOCA Events: Engineering data are needed on fuel behavior, release of fission products from fuel, and thermal hydraulic behavior of the core and primary coolant system during anomalous transient and small LOCA events. Accordingly, NRC has the following types of tests and analytical studies under consideration:

(i) Obtain engineering data on heat transfer and coolant flow conditions for transients and small LOCAs from nuclear and nonnuclear test facilities, also data cooling and fuel behavior under natural circulation and transient conditions where the core may be uncovered.

(ii) Investigate coolability of severely damaged fuel resulting from certain transient and small LOCA events, including flow tests on fuel assemblies which have boiled dry. Study the rate and nature of fission products released from damaged fuel, and transport of fission products from primary system to containment.

(iii) Accelerate the development of advanced best estimate codes to analyze a variety of transient and small LOCA events under various failure conditions, including development of (a) fast running, less precise codes for studies of plant behavior and (b) precise codes for benchmarking the fast running codes.

(iv) Develop event trees to define accident sequences leading to extensive core damage.

(b) Enhanced Operator Capability: The need for systems improvement to enhance in-plant accident responses was emphasized in NRC's report to Congress in 1978 on improved safety of LWRs. The needs here include:

(i) research to define improved requirements for data display and diagnostic systems to provide increased assistance to the operator in responding to an accident, including data transmission and communication for assistance and advice by outside organizations.

(ii) Improvements are needed in instrumentation to measure plant conditions such as valve position indicators and reactor vessel water level. Studies should be performed to define all instruments needed to assist plant operators in the diagnosis of accident conditions, and tests should be conducted to evaluate and improve reliability of such instrumentation under long-term accident environments.

(iii) Requirements should also be developed to improve the use of simulators in studying operator response to accident situations and for related training. Such requirements should include consideration of accidents which go beyond engineering failures defined in design basis accidents. Control room and plant protection system design requirements should also be studied to define improvements which will enhance accident response and reduce the likelihood that a plant can operate when safety systems are not all operational. Mechanisms which would preclude plant operation under certain conditions should be further defined.

(c) Plant Response Under Accident Conditions: Research is required on response of safety systems and components during accident conditions, and on ongoing physical processes that can lead to further system failures. Such research includes studies on:

(i) primary coolant sampling methods and related coolant chemistry for analysis of amount of failed fuel in the coolant in an accident involving failed fuel.

(ii) the formation of hydrogen gas during an accident, its transport behavior throughout the primary system and containment, and methods for reduction of its concentration in the primary system and containment to decrease the probability of explosion or fire.

(iii) pressure vessel integrity under thermal shock conditions (cold water on hot vessel) at higher pressures, representative of transient and small-break LOCAs, to determine the potential for vessel failure. Previous tests of this nature were performed at lower pressures more representative of large break LOCAs.

(iv) development of requirements for testing of critical plant equipment, pumps, valves, etc. to determine their reliability of operation under severe accident conditions.

(d) Postmortem Examination and Plant Recovery: Postmortem examination of the TMI core, plant components and containment will clearly be very useful in providing data on fuel behavior, fission product transport and plateout and component operability under prolonged accident environments, and in defining plant recovery requirements and risks.

With reference to the additional research needs defined as a result of the TMI accident, the Office of Nuclear Regulatory Research reoriented about $12 M of its FY-79 program, and, for FY-80, has requested a supplemental funding of about $24 M in addition to a reorientation of program funds of about $34 M.

Integrated Reliability Evaluation Program

The TMI accident has indicated the need to apply the safety engineering insights and techniques developed in the Reactor Safety Study to help determine improvements that may be required for the safety of nuclear power plants. In addressing this need, the Office of Nuclear Regulatory Research is initiating a new research task, the Integrated Reliability Evaluation Program (IREP) to (1) identify those nuclear power plants that appear to have a higher level of public risk due to potential accidents than that indicated in WASH-1400, and (2) enhance the capability of the NRC regulatory and probabilistic assessment staff and related contractors in the use of quantitative risk assessment techniques to provide for more rational implementation of the NRC regulatory processes.

The need for the IREP program arises from our recently acquired knowledge that there are some reactors which can have outliers with respect to the engineering risk perceptions gained from the two reactors studied in WASH-1400. Those accident sequences which contributed significantly to the probability of occurrences of various magnitudes of radioactive releases predicted in WASH-1400 give an engineering perception of those factors in reactors that contribute significantly to risks. We now know that the TMI-2 reactor had an accident sequence that raised the probability of accidents significantly above those predicted in WASH-1400. In this connection, NRC has issued orders to U.S. plants of the TMI type requiring that provision be made for inclusion of anticipatory scrams on feedwater transients and for the raising of the set point on the pressurizer power-operated relief valve. We have also performed a reliability assessment of auxiliary feedwater systems in most operating PWRs and found a significant number of systems requiring improvements in reliability. Here also, NRC is requiring the operators of such plants to initiate necessary changes for improving the reliability of the auxiliary feedwater systems. It is clear that the IREP program is needed to identify other risk outliers that may exist.

IREP will involve the development of event trees to identify potential core-damaging reactor accident sequences for all of our reactors in operation or near to operation (except where they are not significantly different). This will be done by constructing system logic models for the applicable systems which affect the course of accidents and by identifying those accident sequences which contribute most to accident risks for each of the reactors. This effort will also include assessment of human factors. The results, presented in the form of probability of release of various amounts of radioactive materials to the environment, will be compared with the equivalent analyses in WASH-1400, and on the basis of such comparison suggestions will be developed regarding changes that might be needed.

It must be recognized that the IREP exercise will be less complete than WASH-1400. A balance has been drawn between the need for completeness and the urgency involved in finding the most obvious risk outliers, while at the same time developing a cadre skilled in the application of quantitative risk assessment techniques. The event trees to be constructed under IREP will be at least as complete as those in WASH-1400 and, since this is the key element involved developing the skilled cadre, there will be no short cuts in this area.

On the other hand, the development of system fault trees as complete as those used in WASH-1400 is extremely time consuming, and their use would not be consistent with the urgency of the task. Therefore, simplified fault trees will be used. This means that while various areas, such as common cause failures, human errors, and some system interactions, will be addressed in IREP, they will not be covered completely. Nevertheless, obvious outliers will be found by this approach and a foundation will be set for future studies if they are determined to be necessary.

Conclusion

In our view the Commission research program has been making excellent progress in meeting its objectives. There is a large quantity of results being produced which are proving useful in helping to carry out our regulatory and licensing activities. The lessons learned from TMI have been of high value in pointing out those areas of research requiring significantly increased emphasis, including work on the development and application of risk assessment techniques.

In this connection, it can be said that within the last few years the role that risk assessment methodology can play in the regulatory process has become better appreciated and understood within the NRC. RES has recommended for several years that increased use be made of these techniques to make our regulatory process more rational, but that such use should be made with due caution. It is felt, for example, that while the performance of a complete WASH-1400 risk analysis on every reactor to be licensed would be an inappropriate application of these methods, their use as one of a number of tools available to decision makers for obtaining valuable insights on nuclear safety would be most effective. Along these lines, one of the recommendations of the Lewis Report was that, "Fault tree/event tree analyses should be among the principal means used to deal with generic safety issues, to formulate new regulatory requirements, to assess and revalidate existing regulatory requirements, and to evaluate new designs."

In its review of the TMI accident, the Kemeny report criticized the NRC for not having made systematic use of WASH-1400 in its design review analyses, observing that, "WASH-1400 showed that small-break LOCAs similar in size to the accident at TMI were much more likely to occur than the design basis large-break LOCAs, and can lead to the same consequences. Further, the probability of occurrence of an accident like that at Three Mile Island was high enough, based on WASH-1400, that since there had been more than 400 reactor years of nuclear power plant operation in the United States, such an accident should have been expected during that period."

Thus, in the aftermath of TMI, it is clear that the enhanced use of quantitative risk assessment techniques may be expected to be significantly accelerated within the U.S., and perhaps elsewhere, as indicated by the increasingly extensive work in the field now underway in a number of countries of the CSNI. In this development, our experience underscores the importance of having on hand a skilled cadre of practitioners who can use these methods properly. Our IREP program is designed to provide the U.S. with the beginning of such a cadre.

FIGURE 1

ORIGINS OF THE OFFICE OF NUCLEAR REGULATORY RESEARCH

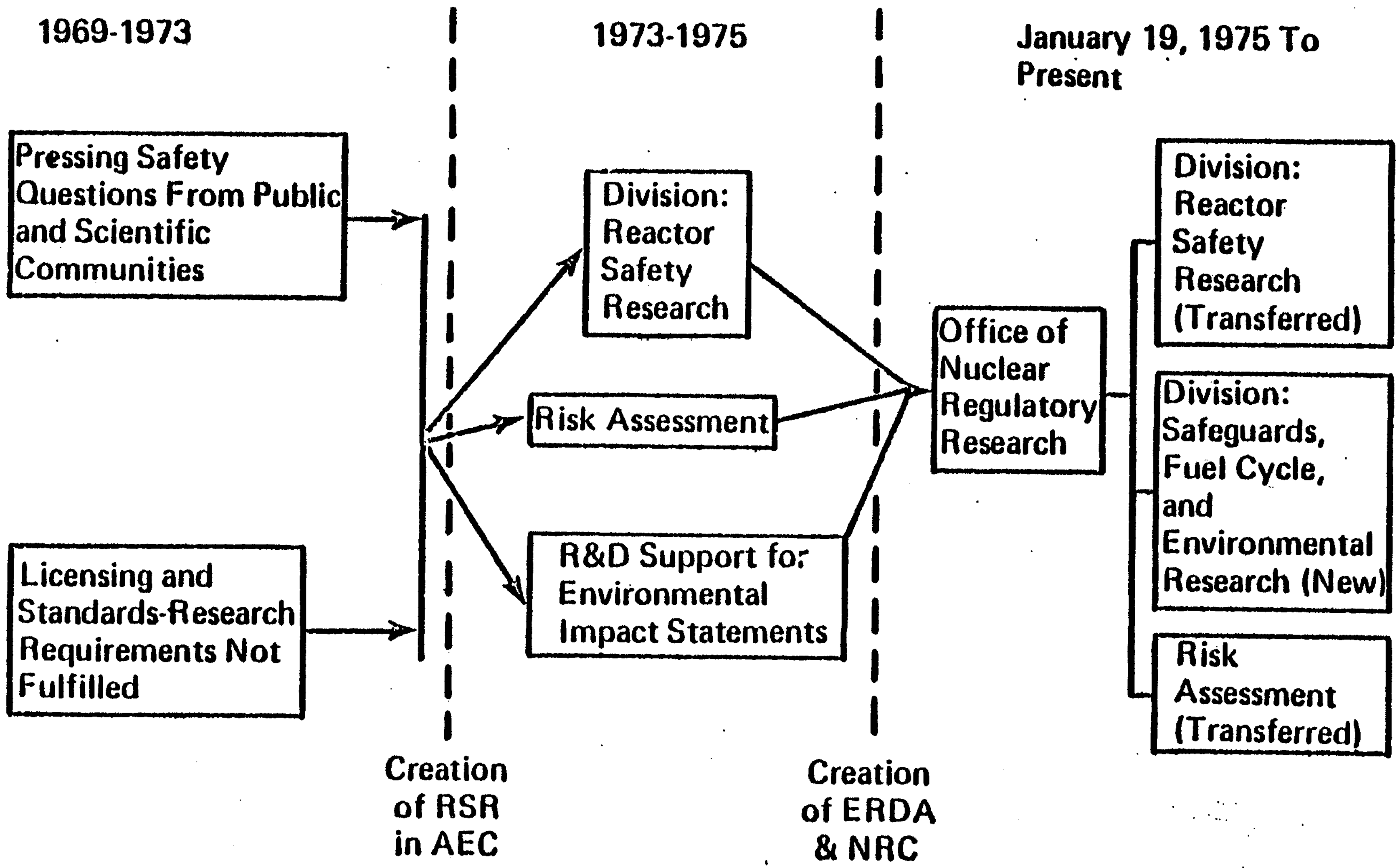


FIGURE 2

RELATIONSHIP OF CONFIRMATORY RESEARCH TO NRC FUNCTIONS

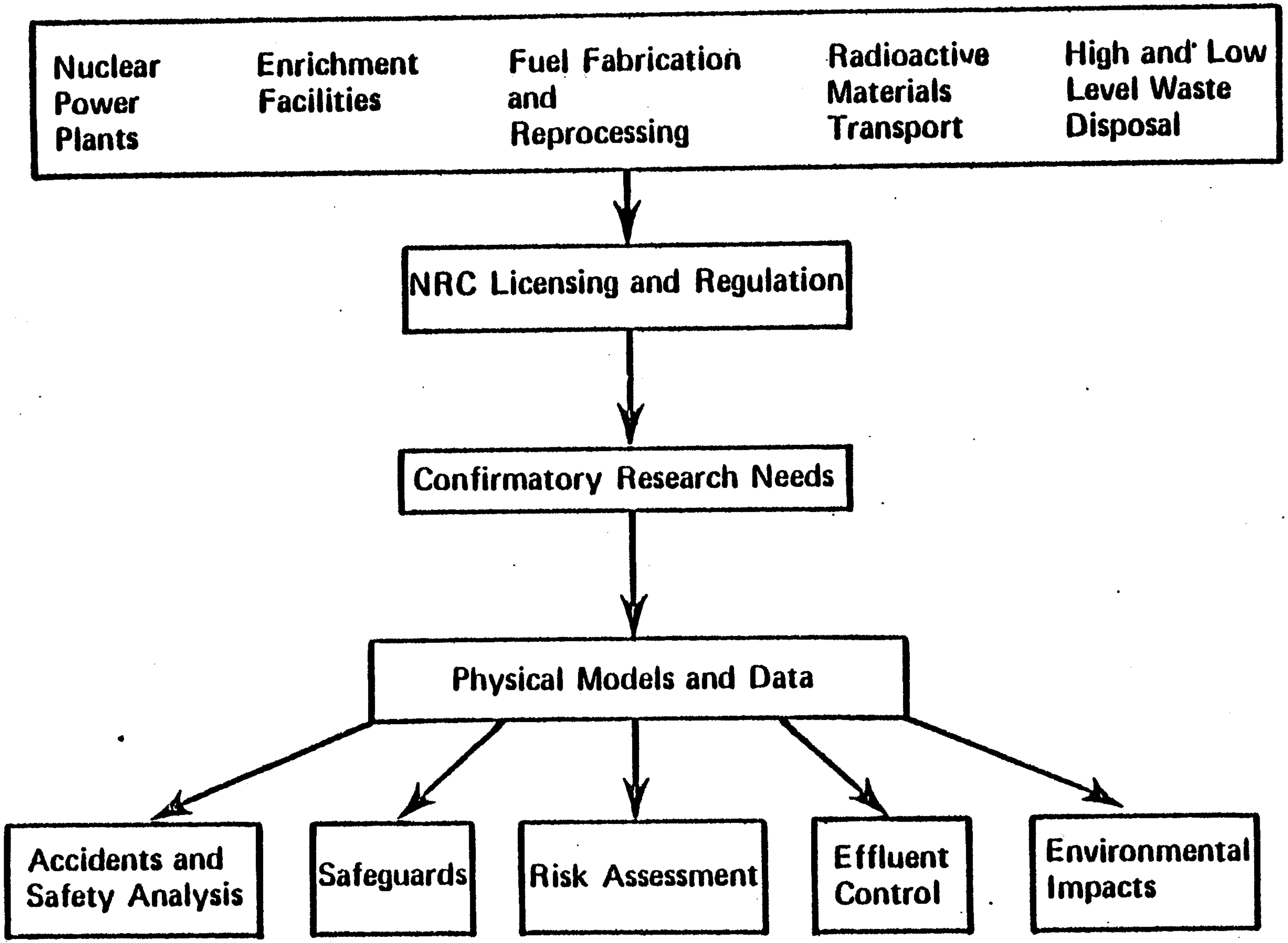


FIGURE 3

USER REQUIREMENTS FOR REGULATORY RESEARCH

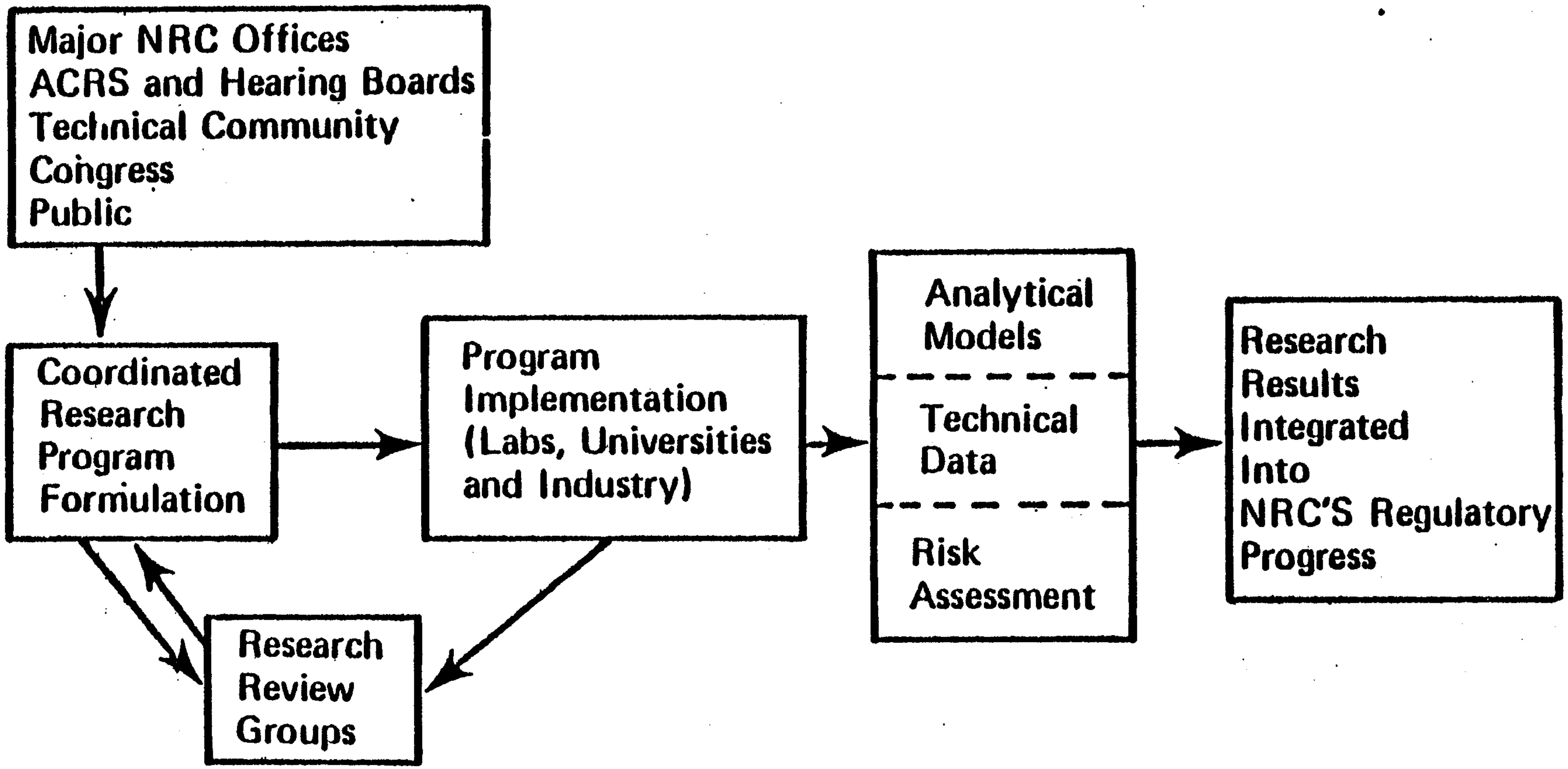


FIGURE 4

OBJECTIVES OF RES PROGRAM

- To Maintain a Research Program Which Supports Assurance of Public Health and Safety and Public Confidence in the Regulatory Program.
- To Provide Objectively Verified Safety Data and Analytical Methods Which Meet the Needs of Regulatory Activities.
- To Provide Better Quantified Estimates of the Margins of Safety for Reactor Systems, Fuel Cycle Facilities, and Transportation Systems.

  Program is Largely Directed Toward Defining Safety Margins. In many Areas, NRC Licenses Have Been Granted on the Basis of Conservative Safety Margins to Account for Lack of Information. A Better Quantitative Definition of These Margins is Required.

- To Establish a Broad and Coherent Exchange of Safety Research Information With Other Federal Agencies, Industry, and Foreign Groups.

FIGURE 5

NUCLEAR REGULATORY RESEARCH PROGRAM

REACTOR SAFETY

LIGHT WATER REACTORS

- LOSS OF COOLANT ECCS RESEARCH
- FUEL BEHAVIOR
- PRIMARY SYSTEMS INTEGRITY

GENERAL REACTOR SAFETY RESEARCH

- SEISMIC ENGINEERING
- STRUCTURAL MECHANICAL ENGINEERING
- SITE SAFETY

ADVANCED REACTORS

- LIQUID METAL BREEDER REACTOR
- GAS COOLED REACTORS

SAFER RESEARCH

- WASTE MANAGEMENT
- ENVIRONMENTAL RESEARCH
- FUEL CYCLE SAFETY
- SAFEGUARDS

RISK ASSESSMENT

- METHODS DEVELOPMENT
- APPLICATIONS OF TECHNIQUES
- IMPROVED SAFETY RESEARCH

FIGURE 6

RESEARCH PLAN FOR CONFIRMED LWR SAFETY ANALYSIS METHODS

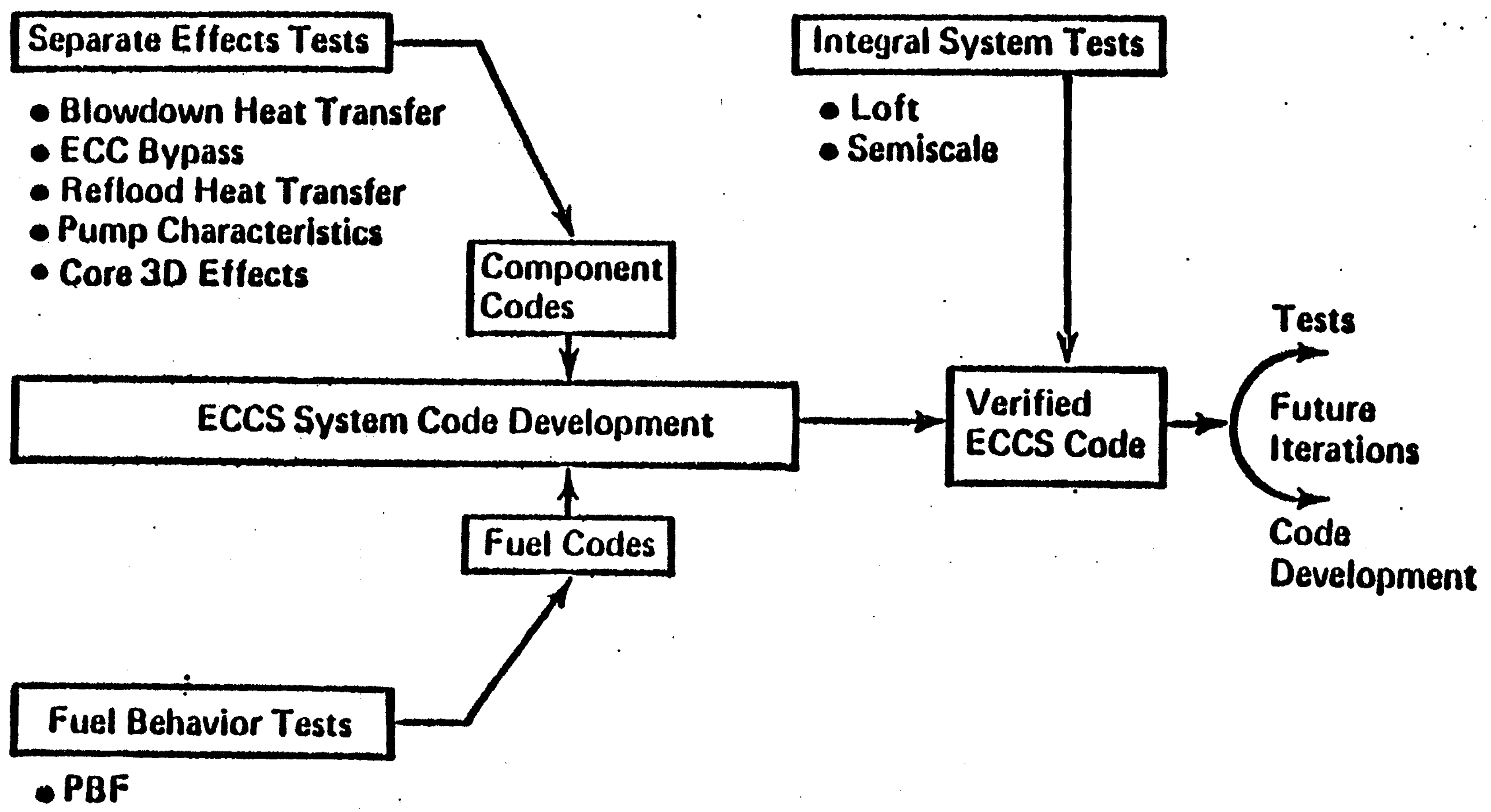


FIGURE 7

PWR LOCA AUDIT ANALYSIS WITH WRAP CODE PACKAGE

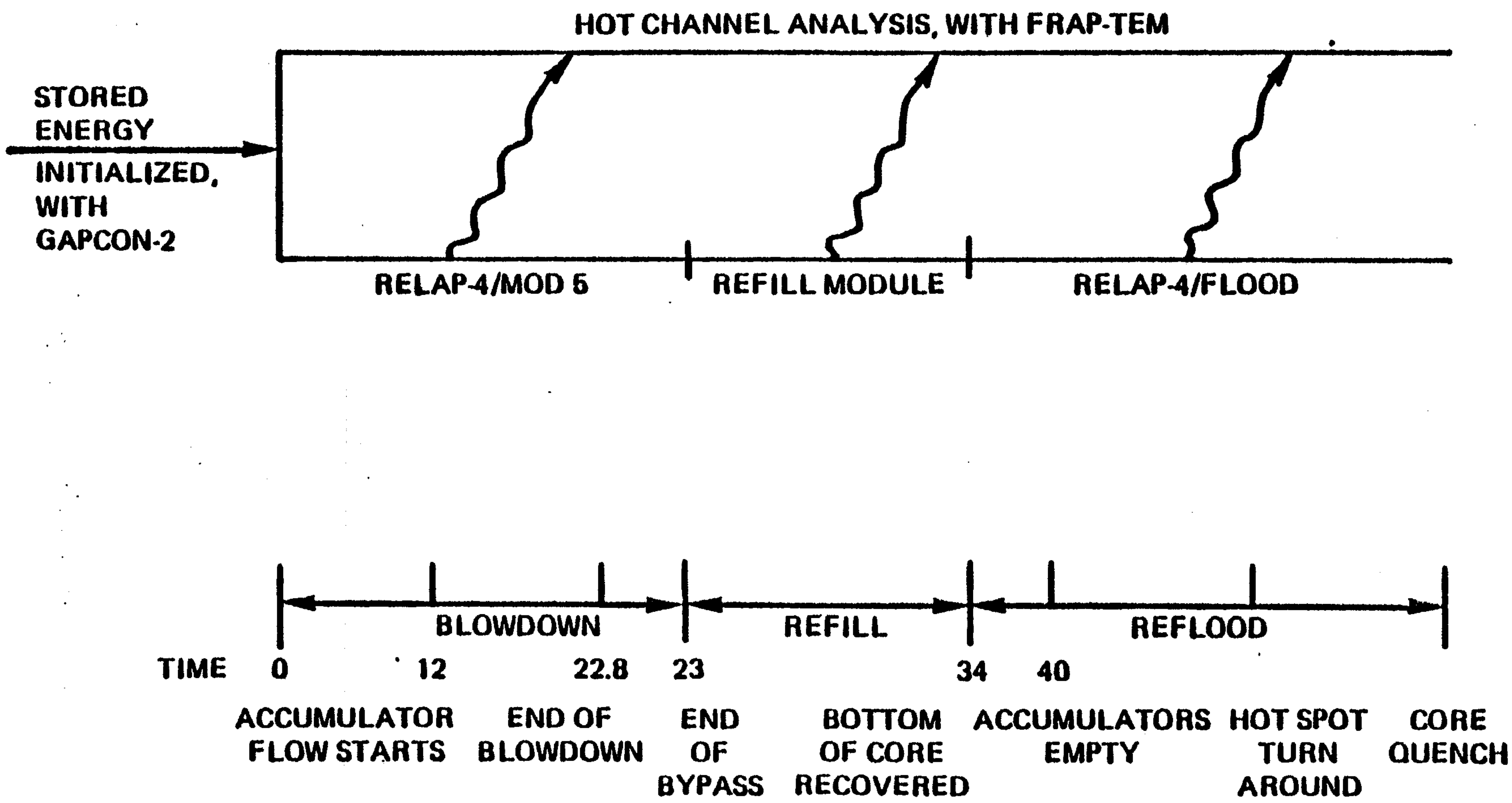
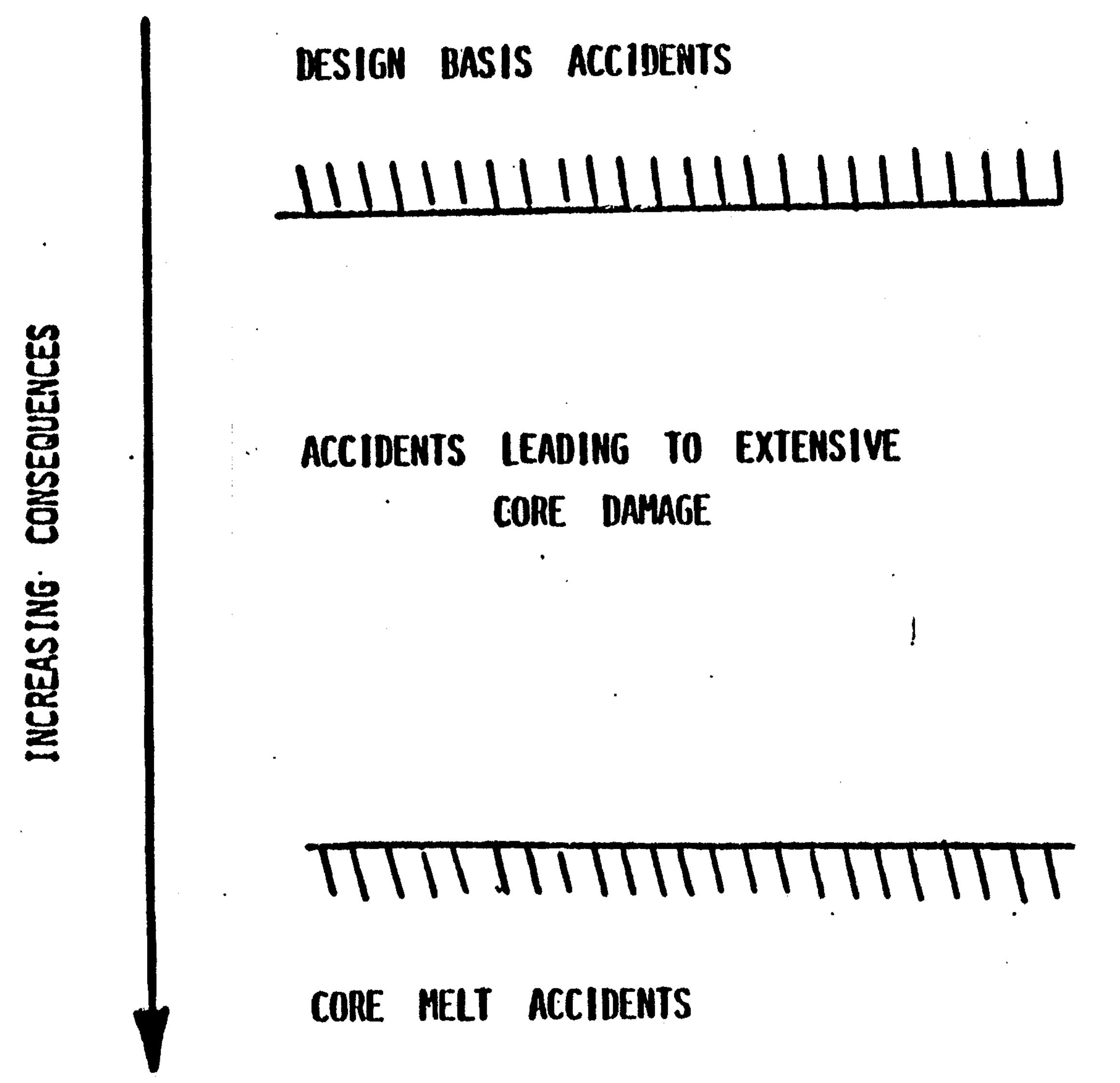


FIGURE 8


INCREASING CONSEQUENCES
DESIGN BASIS ACCIDENTS
ACCIDENTS LEADING TO EXTENSIVE CORE DAMAGE
CORE MELT ACCIDENTS

APPENDIX

LIST OF NRC RESEARCH INFORMATION LETTERS

RIL No.	Date Issued	Subject
1	03/74	ORNL V-5 Intermediate Vessel Test Result
2	05/74	Seismotectonic Map of the Eastern United States
3	08/74	V-7 Intermediate Vessel Test Result
4	09/74	Map Showing Recency of Faulting in Coastal Southern California
5	06/76	Confirmatory Pressure Vessel Test Under Pneumatic Loading
6	10/76	Draft Report: "A Critique of the Board-Hall Model for Thermal Detonations in the UO_2-NA System
7	08/76	The Simmer Code for Analysis of Hypothetical Core Disruptive Accidents in LMFBRs
8	01/77	Decay Heat Data Applicable to LOCA Evaluation
9	03/77	High Temperature Oxidation of Zircaloy Fuel Cladding in Steam
10	02/77	Pressure Vessel Failure Probability Prediction
11	09/77	IEEE Nuclear Reliability Data Manual
12	06/77	Modifications to Pressure Vessel Failure Probability Prediction
13	11/77	Residual Stresses in Welds
14	11/77	Physical Separation Criteria for Electrical Cable Trays (Horizontal Open Space Configuration)
15	12/77	Characterization of BWR Feedwater Nozzle Corner Cracks
16	12/77	Warm Prestressing
17	05/78	Power Burst Facility (PBF) Single Rod-Power-Cooling Mismatch (PCM) Test Results
18	11/77	Frantic Computer Code
19	01/78	GO Methodology Assessment
20	01/78	A Study of Physical Protection Equipment
21	03/78	Critical Review of Sodium Hydroxide Aerosol Toxicity
22	09/78	"MCSS" Model for MC&A Evaluation
23	04/78	EASI Adversary Sequence Evaluation Model
24	04/78	FESEM Adversary Sequence Evaluation Model
25	03/78	FRAP-3
26	04/78	The Impact of Offshore Nuclear Generating Stations on Recreational Behavior at Adjacent Coastal Sites
27	06/78	BEACON/MOD 2
28	05/78	MELT/CONCRET Interactions
29	06/78	Fuel Rod Analysis Computer Code: FRAP-T3
30	06/78	Barrier Penetration Data Base; Physical Protection Assessments
31	07/78	Assay of Standard Reference Material (SRM) 950b
32	08/78	Improvements in the Aerosol Behavior Code for Radiological Assessments of LMFBRs
33	08/78	Plutonium Accident Container Program Research, Design and Development
34	08/78	Nuclear Decay Data for Radionuclides Occurring in Routine Releases from Nuclear Fuel Cycle Facilities

RIL No.	Date Issued	Subject
35	09/78	SFACTOR: A Computer Code for Calculating Dose Equivalent to a Target Organ per Microcurie-Day Residence of a Radionuclide in a Source Organ
36	09/78	Evaluation of General Atomic Codes: Oxide-3, SORS, TAP, and RECA
37	09/78	LOFT Reactor Safety Research Results Through October 1, 1978
38	10/78	Results of the Initial Series of ACPR Experiments on Prompt Burst Energetics with Fresh Oxide Fuel
39	11/78	RELAP-4/MOD 6
40	12/78	The Computer Code BRENDA: A Computer Program for the Dynamic Simulation of a Liquid Metal Fast Breeder Reactor Plant
41	12/78	Laboratory Testing Procedures to Determine the Cyclic Strength of Soils
42	12/78	Critical Experiments Program for Neutronics Code Verification
43	01/79	The Super System Code, A Computer Program for Dynamic Simulation of LMFBR Power Plants
44	01/79	Radiation Dose to Construction Workers at Operating Nuclear Power Plant Sites
45	02/79	The Concept Computer Code and Capital Costs for Boiling Water Reactor Plants
46	03/79	Effectiveness of Cable Tray Coating Materials and Barriers in Retarding the Combustion of Cable Trays Subjected to Exposure Fires and in Preventing Propagation Between Cable Trays
47	03/79	INREM II: A Computer Implementation of Recent Models for Estimating the Dose Equivalent to Organs of Man from an Inhaled or Ingested Radionuclide
48	04/79	A Tectonic Overview of the Midcontinent
49	04/79	In Vitro Dissolution of Uranium Product Samples from Four Uranium Mills
50	04/79	Criticality Safety Guidance
51	04/79	The Concept Computer Code and Capital Costs for Pressurized Water Reactor Plants
52	04/79	Earthquake Intensity Scale
53	05/79	Debris-Bed Coolability Limits, Results from In-Core Tests D-1, D-2 and D-3
54	05/79	The Set Equation Transformation System
55	05/79	The Concept Computer Code and Capital Costs for High and Low Sulfur Coal Plants
56	07/79	Effects of Nuclear Power Plants on Community Growth and Residential Property Values
57	08/79	Small Scale ECC Bypass Research Results
58	08/79	Comparison of Simulation Models Used in Assessing the Effects of Power Plant Induced Mortality on Fish Populations
59	09/79	Transient Fuel Rod Behavior Code: FRAP-T4
60	10/79	Seismicity and Tectonic Relationships of the Nemaha Uplift in Oklahoma
61	10/79	Molten Sodium Interaction with Basalt Concrete
62	10/79	New Madrid Seismotectonic Study

Discussion on Session VII

J. Edwards (UK) (to A. Spano and J.S. MacLeod)
It is salutary to realise that after 25 years of PWR experience, and a great deal of extensive research, the TMI accident has revealed the need for very much more research. This work will take years to complete, and I ask if any one particular aspect of the wide spectrum of research planned and in hand has been selected to be of top priority and urgency.
The ultimate responsibility of safeguards is the protection of the public, and whilst much of the research and analysis is plant orientated, the final protection barrier is the containment, and therefore its integrity under all conditions must be guaranteed. Would not research directed to this end qualify as a strong contender to be at the top of the priority list? And is it at the top of this list?

A.H. Spano (US)
Containment integrity is, of course, a very important aspect in assuring that the risk to the public is low, and one of the projects in the USNRC program related to improved reactor safety deals with the study of alternate containment systems features for their risk reduction potential in mitigating accident consquences. However, as indicated in WASH 1400 and as was demonstrated during the TMI accident, and reiterated in the Kemeny Report, the problem of human failures and of the man-machine interface has not been adequately addressed so far, and appears to deserve the highest priority in the NRC program at this time.

J.S. MacLeod (UK)
The priorities should be prevention of the accident, mitigation of its consequences, and containment. In the UK great emphasis has been placed on containment and for example an extensive study of its behaviour on a time dependent basis has been carried out. It would be expected that the same emphasis would be put in relation to LWRs.

F.W. Heuser (F.R. of Germany) (To A. Spano)
There are at least two levels of interest one has to look at, if one tries to come up with conclusions on risk procedures and risk assessment. The first one, namely the probabilistic approach by event tree and system fault tree analysis, has been emphasised strongly in your presentation, Mr. Spano.
Going beyond the questions of plant oriented analysis, can you make some comments on the 2nd aspect, dealing with further im-

provement of consequence modelling, determination of health effects and overall risk assessment under societal aspects.

A.H. Spano (US)
As is well appreciated, the assessment of risk for postulated releases of radioactivity outside the containment depends critically on the consequence model used. Various schemes for calculating consequences have been developed based on a variety of assumptions and models relating to the site, population distribution, dose-health effects, etc. In view of the importance of trying to pin down some of the uncertainty related to consequence modelling, the USNRC is organizing a workshop in the field, to be held at Oak Ridge in 1980, and at the CSNI meeting next week we will propose that this workshop be conducted under the sponsorship of CSNI.

F.J. Turvey (Ireland) (To A. Spano)
You say that the NRC is turning its attention to accidents which lie between the Design Basis Accident and the Meltdown Accident. Do you have specific intentions at present to examine fuel and cladding behaviour at conditions which occur between these two accidents?

A.H. Spano (US)
Yes we believe it is important to give extensive study to small LOCA or transient accident sequence having potential for severe core damage. This means doing in-pile and out-of-pile tests to study fuel behaviour under severe coolability conditions, for example, the limits of coolability of the core under natural circulation conditions as obtained in the TMI accident, or the coolability of damaged fuel in a debris bed configuration. While we will be doing some work with PBF, we recognize that because of the limitations of the test facility, in particular with respect to fuel length and bundle size, consideration may have to be given for doing these important tests elsewhere in order to get a correct understanding of the thermal hydraulics and fuel damaging phenomena involved.

C. Ringot (France)
En France, nous préparons actuellement un nouveau programme lié à TMI pour étudier le comportement des éléments combustibles dans l'hypothèse d'un situation de fonctionnement à température élevée pendant des durées longues (de l'ordre d'une heure ou plus).
Ce programme comprend trois étages:
(1) Une étape analytique d'étude de comportement des gaines dans des gammes de températures allant de 600°C à 1000°C en présence de vapeur d'eau;

(2) Des essais sur crayons dans les mêmes conditions;
(3) Une étape de comportement global dans le réacteur PHEBUS sur l'assemblage qui fera suite au programme actuel qui a trait au comportement des éléments combustibles lors d'une grande brèche (LOCA). Je rappelle que cet accident est l'accident de dimensionnement des réacteurs à eau pressurisée qui est toujours d'actualité même après TMI.

N. Aybers (Turkey) (To A. Spano)
I would like to ask how the existing safety codes are in agreement with the data obtained from the TMI accident.

A.H. Spano (US)
We are making many calculations to try to understand some of the phenomena we observed in the TMI accident, including fuel calculation in high temperature regions where our ECCS codes have not previously been applied. We are planning to extend our codes to cover fuel behaviour in these high temperature regions, the loss of coolant geometry, and the effects of non-condensible gases.

SESSION VIII

SPECIFIC ASPECTS OF REGULATORY REVIEW

Chairman: J. Van Daatselaar

Scientific Secretary: A. Jensen

BACKFITTING

Dennis M. Crutchfield
Division of Operating Reactors
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission

The Nuclear Regulatory Commission, and its predecessor, the Atomic Energy Commission, has always had an active program to improve reactor safety. Operating nuclear power plants are not insulated from further safety improvements. Continuing improvements to existing plants are made based on operating experience, and new knowledge or understanding of safety issues through research, testing, and analysis. Such improvements are frequently referred to as "backfitting".

Methods of Backfitting

New or revised regulations and generic letters are the two primary methods for achieving facility modifications or procedural changes to provide substantial additional protection for public health and safety.

Assessments of safety issues that arise may lead to a determination by the staff of the need for basic safety requirements. These are listed in the Commission regulations as set forth in Title 10 of the Code of Federal Regulations (10 CFR). Over the years many new requirements important to safety have been issued. Examples of new requirements are:

1. Quality Assurance Criteria, 10 CFR Part 50, Appendix B, June 1970.

2. Codes and Standards, 10 CFR 50.55a, June 1971.

Paper prepared for presentation at the CSNI Specialist Meeting on Regulatory Review in the Licensing Process, Madrid, November 7-9, 1979.

3. Fracture Toughness Requirements, 10 CFR Part 50, Appendix G, July 1973.

4. Acceptance Criteria for Emergency Core Cooling Systems, 10 CFR 50.46, January 1974.

5. As Low As Is Reasonably Achievable - Reactor Effluents, 10 CFR, Part 50, Appendix I, May 1975.

6. Physical Protection Against Industrial Sabotage, 10 CFR 73.55, February 1977.

The NRC regulations also include a specific section (10 CFR 50.109) which states that matters can be required to be backfit when that change will provide substantial additional protection that is required for the protection of the health and safety of the public or the common defense or security.

Many safety issues that arise are assessed for applicability to each operating plant, and as appropriate, a generic letter describing the issue and desired action is sent to affected licensees. When a review shows that the facility does not meet present day acceptance criteria, the licensee is likely to propose changes that improve plant safety. Such changes may fall short of the requirements for new facilities. Over the years this practice has led to many plant improvements by upgrading plant systems through a program of retrofitting that is less formal than using the provisions of 10 CFR 50.109. However, as a result of the identification of certain safety issues through utility/owner groups (working with the staff) or through problems identified as a result of operating experience, Bulletins and Orders have been issued to all operating plants to appraise them of the safety concern and to ensure that all plants address the issue and perform necessary modifications. Consequently, to date the need to invoke 10 CFR 50.109 has not arisen. Often these additional requirements are the result of a new interpretation of existing regulations and not as the result of a new regulation. Examples of less formal upgrading are:

1. BWR Channel Box Wear
2. Fire Protection
3. Mark I Containments - BWR
4. Reactor Vessel Pressure Transients - PWR
5. Pipe Support Base Plate Designs

Assessment For Backfitting

In 1974 our Regulatory Requirements Review Committee was established. This group, composed of senior technical management personnel, is chartered to review significant newly proposed regulatory requirements or proposed changes that provide significant relief from existing requirements, and decide whether, when, and to what reactor plant these changes should be applied. The Committee provides assurance that backfitting (both procedural and hardware) is controlled and justified. Thus, the committee will provide final management review of all those items which constitute significant changes from or additions to existing Regulatory Requirements and interpretations.

Problems arising at operating reactors may require immediate action. Such issues are generally handled within the framework of existing licensing requirements. Requests of licensees for additional information may be done by generic inquiry by the Office of Inspection and Enforcement or by the Office of Nuclear Reactor Regulation. Resolution of these issues typically include one of the following:

1. Equipment modifications;
2. Changes to plant procedures or operating modes;
3. Changes to technical specifications - under existing requirements; or
4. A new safety requirement that must be approved by the Regulatory Requirements Review Committee with attendant changes to technical specifications.

The Committee assesses each proposed new licensing requirement. The proposal may stem from staff safety analyses, research, staff-industry assessments conducted as codes and standards working groups, or from operating experience. The Committee characterizes each new licensing requirement into one of the following categroies:

1. Forward fit only, no need for further staff consideration of application to older plants;
2. Further staff consideration of the need for backfit should be carried out, on a case-by-case assessment; or,
3. Application to all facilities required.

Programs Related to Backfitting

Staff programs underway that will entail backfitting on several issues to varying degress may be of particular interest to you. These rather extensive efforts are the Systematic Evaluation Program and the short term safety-related recommendations of the Three Mile Island Task Force.

Systematic Evaluation Program

Because of both the evolutionary nature of licensing requirements and technology developments over the years, operating nuclear plants include a very broad spectrum of design features depending upon when the plant was constructed and licensed for operation. (The oldest operating plant was licensed in 1959). Documentation definitively describing safety design characteristics is, in general, correlated to the age of the plant - the older the plant the less complete the documentation. Also, the older the plant the more likely it is to be at variance with some current licensing requirements; although there has been substantial modifications made as a result of routine licensing reviews.

About 1-1/2 years ago the implementation phase of the Systematic Evaluation Program was initiated to assess the safety adequacy of 11 of the older operating plants. This program is engaged in a detailed review of these plants as they compare to present NRC requirements for new plants. The technical evaluation will be based on the assessment of more than 130 selected safety topics. Each topic review will document the extent to which it conforms to current licensing criteria and specifically identify any deviations. The acceptability of deviations from current criteria will be determined based upon an assessment of the reduction in safety margin and the availability of non-safety systems to perform or substitute for safety grade equipment. Exemptions to current criteria will be considered if an alternate acceptable method for accomlishing the function is available. In the absence of acceptable alternatives selective backfitting will be required ranging from procedural or technical specification changes to hardware modifications.

The SEP review is completed by performing an integrated assessment by considering all the identified unacceptable deviations. The intent of the integrated assessment is to optimize the benefits from any required modifications ensuring that the impact of these modifications are minimized while achieving maximum enhancement of plant safety. Immediate backfitting may be required prior to completion of the topic or DBE reviews if a deviation from current criteria is deemed to be of a substantial safety concern.

One of the major topics in the SEP involves seismic design considerations. Seismic design criteria evolved significantly during the period 1956 - 1967 when the 11 SEP facilities received their Construction Permits. Consequently, the seismic designs of these plants vary considerably. The seismic review of the SEP plants is proceeding in two steps. First, currently available information is considered to determine a site specific seismic design input, to be used for establishing the safe shutdown earthquake, for each plant if needed. Then the designs of the plants in relation to the site specific seismic inputs is reviewed.

The engineering judgment of the staff must continue to be the primary measure of the safety significance of each issue for each plant. Consideration will be directed to achieving the required safety objectives with existing plant equipment, other than hardware normally used for the purpose.

With the current budgeting of 34 man years of effort and 1.3 million dollars, the evaluation of the eleven facilities is scheduled for completion by May 1982. At that time NRC management will evaluate the program to determine the appropriate course of action for the remainder of the operating reactors. The decision would be based on the prospects for real improvements to operating safety with impact/value consideration for the remainder of the plants. The management evaluation is expected to determine the most efficient means of proceeding with the remaining reactors or whether there are bases for termination of the program.

The Advisory Committee on Reactor Safeguards (ACRS) has recently communicated its reaffirmation to the Commission of their recognition of the importance of reevaluation of the reactors under review in the SEP and of the importance of developing a suitable process for other reactors. The ACRS also recommended that the NRC reevaluate the structure of the program.

In the same context, one of the recommendations of the Presidential Commission on the accident at Three Mile Island is that the NRC "... conduct systematic reviews of operating plants to assess the need for retroactive application of new safety requirements."

In summary, the need exists and is recognized to maintain a systematic evaluation program for operating plants. Although the structure and format of the SEP may undergo changes over the next several years, the goal of improving safety at operating nuclear facilities will not diminish.

Plant Up-Grading Resulting From TMI-2 Task Force Assessments

Following the incident at TMI-2 all holders of operating licenses were instructed to take a number of immediate actions to avoid repetition of the error made at TMI which contributed to the severity of the event. The instructions were transmitted via bulletins issued by the Commission's Office of Inspection and Enforcement. Licensees of the B&W design reactors shutdown until certain short term actions were completed and reviewed by the NRC staff. In addition to the short term modifications, the licensees proposed to implement certain additional long term modifications to further enhance the capability and reliability of the reactor to respond to various transient events. These actions were confirmed by a Commission order to each licensee.

The NRC Lessons Learned Task Force identified several safety concerns that should require near-term licensing considerations for both operating reactors and facilities under licensing review. These items are discussed in NUREG-0578.

Following discussions with the Commission, the Regulatory Requirements Review Committee, and the Advisory Committee on Reactor Safeguards, the staff has identified several areas pertaining to plant design and analysis, and plant operations that require upgrading that are consistent with existing NRC regulations. The implementation of these actions, as outlined below, are to be completed by 1981.

1. Emergency Power Requirements to Enhance Primary System Pressure Control
2. Full Scale Testing of Relief and Safety Valves
3. Control Room Indication of Position of Relief and Safety Valves, and Detection of Inadequate Core Cooling
4. Improved Containment Isolation
5. Dedicated Containment Penetrations for Plants with External Hydrogen Recombiners
6. Improved Radiation Leakage Integrity and Personnel Shielding of Systems Outside Containment
7. Improved Reliability of Auxiliary Feedwater Systems
8. Improved Analyses and Training of Operators for Transients and Accidents
9. Provisions for Venting Gases from Primary Coolant System
10. Improved Reactor Operations Command Functions
11. Improved In-Plant Emergency Procedures

Other items identified as a result of the Task Force's assessments that require either rulemaking (new regulations) or further staff assessment are: that make unavailable equipment having selected safety functions; (2) requirements to inert Mark I and II containments; (3) requirement for all reactors to have a hydrogen recombiner system; and (4) requirement for an on-shift Technical Advisor.

Concluding Remarks

One of the basic findings of the Kemeny Commission was that one must continually question whether the safety features already in place are sufficient to prevent major accidents. The programs we have that are already in place for backfitting and the systematic evaluation of the operating plants will, I expect, receive additional emphasis in the near future. The programs along with expected changes in the licensing process should provide added confidence that nuclear power can be operated with adequate protection for the public health and safety.

First Period of Discussions on Session VIII

R. Gausden (UK) (To D. Crutchfield)
The paper states that 34 man years of effort and 1.3 million dollars are to be expended on the evaluation of 11 facilities Obviously this will not allow a complete re-assessment of each plant - can you give some indication how you are allocating effort and on what basis.

D. Crutchfield (US)
The resources allocated to the program are applied to each of the approximately 130 topics in accordance with previous experience based on our C P and OL reviews. We know about how much effort it takes to review each of these items on a current licensing application and can estimate the effort for older operating plants.

INSPECTION-EVALUATION INTERPHASE IN THE DESIGN AND CONSTRUCTION OF NUCLEAR POWER PLANTS

Luis Santomá Juncadella
Inspection Operative Unit
Board of Nuclear Energy (JEN) Madrid

This work analyses some questions to be considered in the planning of inspection-evaluation interface. Some characteristics of a nuclear project are analysed, such as its importance relative to safety or their frequency. Also analysed are the types of reviews and controls used by the various participants in a project. Finally, the types of tests to be performed by regulatory agencies are classified.

1. Introduction

This work presents the group of problems in the interfaces existing between inspection and evaluation activities performed in the field of nuclear safety by a regulatory agency.

Traditionally, the concept of inspection has been associated with direct testing of components, systems and structures in plants or in the site of nuclear power plants, while, on the other hand, the concept of evaluation has been associated with the analysis or review of documents, drawings and calculations in an office. Nevertheless, this concepts are not in actual practice so clear-cut, since inspectors conduct physical inspections that have to be compared with detail drawings and it is obvious that it would be impractical for evaluators to examine all and every of the detail plans generated during the project.

The design and construction of a nuclear power plant are in fact parts of a continuous process starting from the basic design and ending with the startup tests, and as the common goal for inspectors and evaluators is to assure a safe and reliable nuclear plant, the tasks of both groups must be determined so as to better ensure the attaining of such common goal and this leads, as we will see, to the need of a close working relationship in some project stages between inspectors and evaluators, that is, an interphase establishment.

It is helpful, in order to establish such interphase, to analyse from the beginning the different activities involved in the design and construction of a nuclear plant without overlooking any single stage or work area. It is also necessary to consider the particular aspects of the major participants, as well as the type of control to be used on the different activities. Lastly, the very regulatory agency will have to plan a control and supervision strategy in view of the above, taking into account its own possibilities and those vested upon it by the legislation existing in the country. We would like to set out that, starting from the experienced gathered in previous projects, it is possible to carry out the above mentioned analysis within a relatively short span of time, deriving consequently, an improved efficiency of the supervision and control tasks.

2. Analysis of the Activities in a Nuclear Project

We can identify, within the nuclear safety related activities, which are an integral part of designing and building a nuclear plant, those affecting fundamentally a given system or structure and those affecting an assembly of systems.

The sequence of activities can be clearly established for the first case:

Design

- BASIC DESIGN.
- DETERMINING COMPONENT DESIGN PARAMETERS AND PREPARING TECHNICAL SPECIFICATIONS.
- DETAIL DESIGNING OF COMPONENTS AND SYSTEMS.

Production

- MANUFACTURING.
- SHIPPING AND STORAGE.
- ASSEMBLY AND CONSTRUCTION.
- STARTUP TESTS

As this is a sequence of activities, nuclear safety will be compromised at each stage by the previous stages. Thus, per instance, it is evident that a high quality manufacturing does not imply a high nuclear safety in the system if this has not been correctly designed and it is necessary, therefore, that all and every of these stages be performed at a consistent level of quality.

The second case, that is, activities simultaneously related to various systems is represented by studies and reports such as: accident analyses, determining containment parameters, shield calculation, missile studies, etc.

When analysing the several activities in a project, it is convenient to do so under the following considerations:

a) According to its importance relative to nuclear safety and to other aspects such as the commercial operation of the plant or the interests of insurance companies:

- Activities almost exclusively related to the commercial operation of the plant as is the case of the turbine and condenser in a PWR. Logically these do not attract the attention of regulatory agencies.

- Activities related at the same time with commercial operation and nuclear safety, such as the mechanical thermo-hydraulic design of the core and primary circuit, the protection and control systems and the mechanical and hydraulic design, manufacture and assembly of residual heat removal systems, of chemical and volume control and of components cooling systems.

  These systems need to be analysed and inspected by the regulatory agency. In many cases, the plant operator would apply strict controls on these activities and the regulatory agency will then be able to make analysis and inspection a little lighter.

- Activities related to nuclear safety and to the interests of the insuring companies, such as fire-protection systems.

- Activities only related to nuclear safety that can either be tested or which may be subject to penalties for proven non-conformity, such as those affecting the containing system, containment sprinklers, safety injection system, shielding calculations and waste treatment system. These activities will be controlled to a lesser degree than the above by the participants in the project.

- Activities concerning nuclear safety, but where the possibility of detecting any failure during the plant life is very unlikely, such as seismic design and testing, flood protection design, accident analysis and determining containment parameters. The regulatory agency must strictly control this type of activities.

b) <u>According to its frequency</u>

- Repetitive, such as hydraulic calculations, supports design, seismic calculations, shielding calculations and structural calculations. Here sampling control techniques may be used.

- Non-repetitive, such as nuclear design, thermo-hydraulic design, containment parameters determination. Here an item-by-item follow up is required.

The type of organisation carrying out the different project activities will have an influence on some aspects of the control performed by the regulatory agency, although there is not room in the present work to go into these details.

Included into such organisations are:

Operating company	Mixed ownership State run Government Research Institutions
Engineering	Domestic Mixed Foreign
Main supplier	Domestic Mixed Foreign

Manufacturers, installers, builders	Domestic Mixed Foreign

3. Supervision and Control of Nuclear Project Activities

The regulatory agency is at the top of a pyramid formed by the several control stages to which the activities for a nuclear project is subject. It is useful to analyse the features of the control structures of the different activities involved, to make possible for the competent authority to select the most suitable form of control in each case.

a) Firstly, the number of organisations intervening in the control of one given activity must be analysed:

- None: This situation can occur during design, and even during manufacture and assembly of some systems and this must be particularly observed by the competent authority.

- One: This situation can occur during the design of many systems where the engineering company checks with its own personnel the compliance of these items, although it does not actually participate in such activities (according to 3rd Criterion, Appendix B of Quality Control, 10 CFR 50). This condition can also occur in some manufacture activities not having been assigned any independent inspection agency.

- Several: A very common condition in nuclear class 1 and 2 mechanical components manufacturing where the manufacturer himself, the Independent Inspection Agency, the Main Supplier and the Plant Owner participate in the control. This situation is not frequent in the design areas, except when the plant owner has its own design review organisation.

b) Secondly, the intensity of the control applied by the different organisations is to be analysed (controlled activity percentage). In some cases, 100% inspections are performed, as is the case with nuclear class 1 weldings, while in other cases the percentage is lower or is performed by statistical sampling techniques, as is the case with the check-up of many detail calculations or with receiving acceptance of materials.

c) In the third place, the type of control used requires consideration, namely: direct control, as performed with non-destructive tests or when a design is recalculated before witnesses during testing, during a systematic cetificate review or during a quality control audit.

d) In the fourth place, account must be taken of the degree of independence of the control organisation relative to the controlled activity. As a rule, the independence increases in the following order: control by the same company charged with the activity, main supplier or engineers, plant operator, independent inspectors.

4. Some Aspects to be Taken into Account in the Preparation of the Strategy Applicable to the Supervision and Control of a Nuclear Project by the Regulatory Agency.

Due consideration to the above aspects, can facilitate the preparation of the strategy applied in the supervision and control of a nuclear project on the part of a regulatory agency, identifying those activities where such control is to be emphasized.

It is worth setting out that the very nuclear legislation of the country may facilitate the control of certain activities. In the Spanish case, per instance, our legislation favours control on site selection and on manufacture and installation of equipment, but conversely it is disadvantageous relative to the detail design control.

Keeping in mind that the main object is to assure that the plant design and construction are within the established nuclear safety limits, emphasis must be placed on the kinds of checks which will more clearly evidence in each case that such objective has been attained. To this effect, several kinds of verifications can be identified, namely:

a) verifications of components or systems easily replaced during plant operation, such as radioactive waste system filters, radiation monitors, etc...

b) verifications that can be carried out during manufacturing in performance tests or during start-up tests. These involve systems and equipment operational aspects.

c) verifications through tests during manufacturing or assembly such as control by non-destructive tests, cable rating, seismic rating, materials testing, etc.

d) verifications through inspections not requiring calculation verification, such as physical separation of electrical systems, location and identification of components, positioning of check valves, etc.

e) verifications requiring a physical inspection followed by calculation checkup, preferably conducted by the same expert. This is the case of the shielding system inspection, piping restrictors and support system inspection, fire protection system inspections, missile protection system inspection, etc...

f) verifications where a physical inspection and calculation verification is possible by two different experts. This is the case of civil construction or structural sizing and piping layout and hydraulic design.

g) verifications not requiring a physical inspection, such as happens with reports and calculation reviewing on matters such as accident analysis, containment parameters calculation, reliability studies, etc...

verification
This classification of types can be used to provide more efficient determination of the work modules that can be assigned to the different experts or specialists of the regulatory organisation.

Follows some of these possible modules, namely:

- Accident analysis
- Detail design of mechanical systems and components
- Civil construction execution
- Manufacture and assembly of mechanical components
- Manufacture and installation of electrical systems, including controls of circuit physical separation
- Shielding detail designing and inspection
- Detail design of hydraulic and thermic calculations
- Supports designing and inspection
- Detail designing and inspection of ventilation and air conditioning systems.
- Detail designing and inspection of fire protection systems
- Civil engineering design
- Seismic design and seismic testing of equipments
- Containment parameters determination
- Etc...

5. Supervision and Control of Activities Connected with the Design and Construction of a Mechanical System

Summarising all the above, we are going to describe next in a simplified way the control and supervision practices used in our country, with reference to the activities of design and construction of a nuclear plant mechanical system.

a) Basic Design

The basic features of a system are defined at this stage, so as to assure that it will be capable to perform its functions within the entire plant design. The established nuclear safety design criteria are applied here, such as redundancies, diversification, insulation, etc. And the type and number of the different components and circuits of the system are incorporated into the design. Flow diagrams are also prepared.

The main supplier and the engineering firms participate in this stage in different percentage levels according to the system involved.

Normally, a basic design control is provided by the main supplier, and this control is more relaxed in the case of engineering firms.

Nuclear safety analysis of the basic design is described in essence in the Preliminary Safety Study, which is analysed by the Evaluation Section.

In this stage, the "reference plant" concept can be used rather successfully in some cases. It is convenient for the Inspection Section to check at this stage:

- whether the project organisations and control assurance general lines are adequate.
- whether applicable codes and standards are defined for each case.
- whether the various systems, components and structures are classified into the appropriate safety and seismic categories.

b) <u>Determination of Component Design Parameters and Preparation of Technical Specifications</u>.

The hydraulic and thermic calculations are performed at this stage to obtain design parameters for the equipment. These calculations will be carried to a finer level during detail designing. Mechanical properties of piping are also determined at this point.

Likewise, purchase technical specifications are prepared at this stage by the main suppliers and engineering firms. Specifications include design requirements, applicable standards, manufacturing techniques, quality tests and assurance and the components to be produced.

Design and specification control at this stage can be in some cases extremely deficient at this stage, while in other cases, a multiple control can be exercised, even with the plant operator participation.

Some of the data and parameters established at this stage are included in the Preliminary Safety Study (EPS), but nevertheless, these data can be interpreted only as guidelines, since they may be later modified during detail designing. In fact, the design situation of the various mechanical systems can vary widely at the time the EPS is prepared.

The analysis of the design parameters obtained at this stage is performed by the Evaluation Section of JEN, while the review of purchase technical specification is the responsibility of the Inspection Section and is performed when the mandatory information supplied by manufacturers, installers and builders is received by this agency.

c) Detail Designing

The detail designing of mechanical systems include the design of components for such systems, the final piping layout, hydraulic and thermic fine calculations, seismic design of supports and restrictors.

As a rule, designing the components is the responsibility of the component manufacturer and must adhere to the design technical specifications prepared by the engineer firm. The control or review of drawings and figures is done by the same engineering firm and, in some cases, by the plant operator. JEN has also the opportunity to control such design, since the drawings are forwarded to this Agency in order to obtain the appropriate manufacturing approval.

Piping layout, thermic and hydraulic calculations, seismic design and support and restrictor design correspond to the engineering companies. The control of these designs can vary widely depending on the circunstances of each case, it can even be non-existing in extreme cases, and this is one of the weak points in the project control. Among the cases contributing to these undesireable situation, we can mention: difficulties in the control of design interfaces, calculation dispersion, scarcity of specialists on certain design areas and the fact that, in many cases, design quality does not affect plant commercial profitability.

Normally, plant detail design is not available to the regulatory agency, save those cases where the preconditions for construction authorisation provide otherwise. It is self-evident that if all detail drawings and calculations were available their review will impose an unsurmountable burden and for this reason the solution has been adopted to make partial, random checks during plant inspections and design office visits, carrying this out jointly by the Inspection and Evaluation staff, some times accompanied by specialists from departments other than Nuclear Safety and even from other organisations outside of JEN.

It is many times convenient to carry out checks in the nuclear plant itself, followed by verifications of calculations and drawings using the Design Quality Control procedures.

d) Manufacture, Shipping, Storage and Assembly

These stages correspond to the physical execution of what has been designed in the previous stages.

Generally, the control of these activities tends to be intensive, with several organisations participating. We can say that this is the best defined and complied with aspect of quality control with principal application of Quality Assurance techniques.

The regulatory agency receives documents on these stages, either directly from manufacturers, builders or installers or through the plant operator. Besides, the presence of independent inspectors is mandatory in many control activities.

On the side of JEN, these stages are controlled by the Inspection Section.

CONCLUSIONS

After analysing the above questions, we can conclude that in the control planning by the regulatory agency in a country with a structure similar to the Spanish structure, it is helpful to:

a) Analyse in their entirety the various activities involved in the project as a whole and their relative importance with respect to nuclear safety.

b) Consider that the design and construction of a nuclear plant is a continuous process and, as a result, there is an inspection-evaluation interphase requiring a structure. Each of the stages in the project must be consistently controlled against the others.

c) The characteristics of the control exercised by the other participating organisations must be considered.

d) The regulatory agency must organise its control actions, taking into account the above described aspects and its own capabilities.

DECOMMISSIONING LICENSING PROCEDURE

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Decommissioning or closure of a nuclear power plant, defined as the fact that takes place from the moment that the plant stops producing for the purpose it was built, is causing preocupation. So this specialist Meeting on Regulatory Review seems to be the right place for presenting and discusing the need of considering the decommissioning in the Safety Analysis Report.

The main goal of this paper related to the licensing procedure is to suggest the need of a new chapter in the Preliminary Safety - Analysis Report (P.S.A.R.) dealing with the decommissioning of the nuclear power plant. Therefore, after a brief introduction the problem is exposed from the point of view of nuclear safety and finally a format of the new chapter is proposed.

1. INTRODUCTION

In spring 1975, it was held in Paris under the auspices of the European Nuclear Society ENS in colaboration - with the American Nuclear Society (ANS), the first world -- conference. This event of such a magnitude in the world was named: "NUCLEAR ENERGY MATURITY".

Soon after this nuclear event we have to be more worried with the next step to maturity, that is, the old - age not of the nuclear energy which will continue its development for the benefit of humanity, but that of the plants which are used as the first source of nuclear energy. The - answer to the question, people generally begin to ask, is specially of interest. It already worries those, more or - less connected with the nuclear industry. The question -- which has already been analized in modern studies, is:

What to do with a nuclear power plant, when it reaches its life time?.

Nowadays, half a thousand nuclear reactors in the world are built or under construction, the second decade of operation in several units overcome, and the definitive -- shut-down of 15 reactors decided. Then, it is necessary to study the problems that brings along the decommissioning - of nuclear power plant. Especially interesting are the safety aspects, the different solutions that the experts - suggest and the measures taken by those who had to close - the commercial or research nuclear power plants. Everything always with the safety assurance that technology - allows at the moment.

The question here asked has already several answers. In a way, there is a consensus of the methods used - when decommissioning a plant, according to the particular circumstances in each case.

At present, we are very near to begin a continuous decommissioning of nuclear power plants. And in the future, it will be more often since nuclear energy is consi dered as the optimum alternative to satisfy the energetic - needs of the 80's and 90's. The importance of the problems is such that a number of international meetings have taken place on this subject. In the last one held in Vienne spon sored by the IAEA, the author presented a paper, reflecting the methods and problems caused by decommissioning up to date. As a consequence it was suggested in that paper - the inclusion of that specific problem in the PSAR. In the technical debate after the presentation, the proposal was considered of rather interest to present it in specifical - meetings of licensing procedure. So this meeting seems to be the right place for presenting and discussin this proposal.

In this paper, we try to pose the problems on this subject, its different options, and the implications it has from the point of view of the nuclear safety. Finally, the importance of the subject and the fact that the implications on nuclear safety will be reduced if the decommissioning pro cedure of the plant has been considered from its design, are reasons why it is proposed to increase the content of the - Preliminary Safety Analysis Report (PSAR) with a specific - chapter dedicated to decommissioning of the plant. This apparently precipitated forecast is not so neither from the - point of view of safety nor of economics as you will see - throughout this paper.

2. PROBLEMS: SAFETY ASPECTS

The generic question here asked needs some precisions before answering it. The lifetime of a nuclear power plant may depend on operation time, economic reasons or other relative interests for the plant or for the licensee, supposed that safety reasons do not lead the regulatory body to set decommissioning.

The "death", decommissioning or closure of a nuclear power plant is defined as the fact that takes place from the moment that the plant stops producing for the aim it was built. It is said producing, as not always the operation of a plant is ceased when technically or economically, it is not adviseable to continue with it in operation.

In spite of the young age of the nuclear industry wherein so much is questioned about safety, the quality assurance in design and construction have so increased that plants have been shut-down rather for economical reasons than for natural age.

Nevertheless, whatever the cause will be that leads to the decommissioning of a nuclear reactor, from the point of view of nuclear safety, it does not suppose the inactivity as there is always the need of some work that, to avoid a potential hazard, requires a special treatment and therefore is costly.

All activity during decommissioning of a nuclear power plant must be done under sufficient control that guarantees the safety of personnel, exposed to it professionally, and of the public in general. To fulfil, a radiological control and a continuous verification of industrial security are necessary. In each case the procedures and standards applicable will be determined. One should anticipate and adjust those cases that require specific operation limitis as a consecuence of a possible abnormal situation.

Like for the construction of a nuclear plant ,a document (PSAR) is required, concerning aspects and systems related to nuclear safety; that is also necessary for the - decommissioning, a preliminary analysis which should be justified before the regulatory body to assess the hazard of - works, concerning dismantling, decontamination and termination.

Concerning the operations involving the existence of gaseous or liquid radioactive effluents, the doses should be lower to the required limits in the regulation in force, and in such a way that the values reached are as low as allowed by the technology of the moment. The volume and activity of the liquid and solid effluents to handle with during the closure of a plant, have been taken into consideration by several studies on the subject. In the report AIF/NESP/0095 several tables are collected giving and ample look on the problem.

The radiological surveillance and physical protection of the site and its enviroments should be a matter of great attention during the closing procedure and transport - until all contaminated and radioactive material is removed.

Let us go briefly through the most frequent me-- thods to shut-down nuclear reactors. Lack of sufficient statistics, prevent to identify common operations determining more or less exact methods to decommission nuclear power - plants. Therefore, for those interested in this subject, - it is preferable to refer to the best known decommissioned reactors. In table I, reasons of its closure and termination are mentioned with its main charasteristics. As you - may see, the majority of them are experimental reactors. In each plant there are so particular circunstances that the most adviseable methods from the point of vien of safety,- does not correspond with the most economic one, neither - with the specific site and plant requirements (nuclear power plant, research centre, university, etc). According to the before said and the existing bibliography relating to this, it is consider as main decommissioning methods for - the major facilities, the four following ones:

TABLE I

CONDITION OF SOME CLOSED NUCLEAR POWER PLANTS

Plant	Site	Characteristics(*)	Reason for shutdown	Methods utilized	Final condition
VALLECITOS	Vallecitos Nuclear Centre	BWR, 10 MWE 1957-63	Technical incident in fuel and high operation cost	Fuel removed and turbine dismantled	Area confined and closed One annual visit
HALLAN	Nebrasca	Graphite sodium 82 MWE ? - 1964	Break blocks moderator (graphite)	Fuel and very contaminated material removed. Stored leftovers in vessel buried in sobsoil	Inaccessible and irrecovable
VALLECITOS	Vallecitos Nuclear Centre	BWR, 20 MWT 1963-65	Finished Tests of nuclear heating	Fuel removed and condemnation of systems with valves sealed	Area confined and closed. Two annual visits.
EL-2	Saclay	Exp. R. of heavy water and Co_2 2 MW 1960-65	Final foreseen experiences	Fuel removed and condemnation of plant.	Final decison pendent but under C.E.A. control
PIQUA	Piquia	Organic 11 MWE 1963-66	Technical reasons of the organic liquid	Confinement of radioactive material in vessel.	Store with three-monthly control
PATHFINDER	Northern States	BWE, 62 MWE 1962-67	Steam separator breaks and high reparation costs according to new safety norms	Fuel removed, pipes sealed and filled with sand. Use of turbines.	Access condemned
CVTR	Carolina	HWR, 17 MWE 1963-67	Loss of technical interest after fuel accident	Evacuation of heavy water. Isolation of valves and not irreversible access but inaccessible for public. Groups to thermal plant.	Annual control Access condemned
G-1	Marcoule	Graphite R. Exp. 1958-68	Termination of experimental and economic operation period.	Fuel removed. Condemnation of plant	Under control of research centre of Marcoule.
BONUS	Puerto Rico	BWR, 17 MWE 1962-68	High maintenance cost	Fuel evacuation Radioactive material hermetically sealed after filling with concrete	The whole as museum Annual control 5 years.
ELK RIVER	Minnesota	BWR, 22 MWE 1964-68	Technical problem. To continue experiences not interesting	Total dismantling	Site cleaning
ENRICO FERMI	Chicago	Quick Na, 60 MWE 1966-71	High experimental programme cost	Na and fuel evacuation. Pipes sealed waiting for solution	Pendent
SEFOR	Arkansas	R. exp. A Na, 20 MWT 1960-72	Termination power of tests on fuel elements at high temperature	Sodium circuits filled with nitrogen and siting fenced in.	Access condemned Annual visit
CHINON	Chinón	Graphite-gas 80 MWE 1962-73	Not economic operation	Fuel removed. Isolation of circuits	Periodical control

(*) Type, power and time operation.

Closing of a nuclear power plant without a possibility of its reoperation. ("mothballing"), that means, the plant - stays in the site but in such manner that, when fuel, radioactive or contaminated liquids are removed, there is no more chance of operation. This implies to take steps to - forbid any access to the plants and a periodical radioprotection control, reducing the minimum number of maintenance to accomplish with safety standards.

Partial closing of the plant, leaving for use the secondary circuit. In this case generally, remains operative the turbine generator set system.

Burying the plant as a tomb ("Entombing"). This consists of the sealing "in situ" within an integrated structure, of the highly radioactive and contaminated components such as pressure vessel and internal components. The final structure should display in a visible place the superficial contamination levels accepted by the Regulatory Body.

Total dismantling of the plant. In this case the site should remain in the state it was before the construction of the - nuclear power plant.

3. DECOMMISSIONING IN THE PRELIMINARY SAFETY ANALYSIS REPORT

Because of the before said and bibliographic references consulted, the safety aspects of ecommissioning nuclear power plants is so important that we consider its inclusion in the Preliminary Safety Analysis Report (PSAR) as necesary:

According to the selected method to shut a nuclear power plant,

- Site will be under a situation of more or less attendance, which may be definitive or just temporal and
- The responsability of the property will not finish at the end of operation (decommissioning date), so that financial cost of decommissioning will be guaranteed, to assure radiological protection to the public.

At firs sight, this previsionmight look premature, but both, from the economical and safety point of view, the proposal is useful for the plant and of great help for the Regulatory Body to fulfil its mission of control and radiological protection of the public in general. Remember the problems in the reactors, whose construction corresponds to an earlier generation and the present standard for the inspection on duty. It just gave rise to a relatively new chapter of the PSAR about quality assurance (Q.A.).

After discussing the advantajes to include a new chapter about the PRELIMINARY PROGRAMME OF THE DECOMMISSIONING OF A NUCLEAR POWER PLANT, let us see what could be a first draft format of this chapter. This format could be taken as a basis for the elaboration of the definitive one, in the case the proposal of including it in the PSAR is accepted.

1. FORMAT AND CONTENT OF THE SPECIFIC CHAPTER OF THE PRELIMINARY SAFETY ANALYSIS REPORT (PSAR) CONCERNING THE DECOMMISSIONING OF NUCLEAR POWER PLANTS.

1.1. Objetive.

To make clear the principal aimed purpose, that - means, previous safety analysis on decommissioning depending on the most probable method according to the specific characteristic of the plant to deal with.

1.2. Methods

To justify the anticipated method by means of the property interest and the requirements of the Regulatory Body.

To the effects redarding the regulatory bodies, - we propose to consider only the following three methods:

METHOD nº 1, corresponding to blocking and sealing "in situ" called "mothballing" in Anglo-Saxon terms.

METHOD nº 2, corresponding to the total burial of the plant, or "entombing".

METHOD nº 3, also named DISMANTLING. In thos case all part of the plant with contamination indications have to be removed.

1.3. Technical and administrative analysis

1.3.1. Nuclear Safety study.
Analysis comparative to the possible decommissioning alternatives.

1.3.1.1. General basis programme.
General criteria, appropiate standards and recommended guides.

1.3.1.2. Decontamination procedures.

In this point the decontamination procedure is - described according to the forecast decommissioning method.

1.3.1.3. Various technics.

Sealing, burial or dismantling technics should be defined. They could be various within the same - decommissioning type.

1.3.1.4. Contaminated material treatment.

- steel
- concrete
- graphite, tec.

1.3.1.5. Material transport.

Though well known for the problems nuclear transport pose in general, it is specially interesting to consider the possibility to requiere very special containers from the dimensional point of view.

1.3.1.6. Scheduled Surveillance.

- Radiological control programme of the personnel involved with the operation.
- Radiological surveillance programme regarding - the followed technic.
- Radiological area surveillance and periodical radiological levels control.

1.3.2. Administrative-economic stydy. General presentation.

1.3.2.1. Human capacity.

- Equipment description. Training.
- Definition of responsabilities.

1.3.2.2. License.

Description of the administrative procedure to follow to apply for the license or allowance to decommission.

(in its time, the application of the decommissioning se to be acompanied with the technical justification necesary according to the method to follow, as well as its concordance with the prior - information presented in the chapter corresponding to the PSAR).

1.3.2.3. Material inventory.

- List of material to decontaminate.
- List of material to transport.
- material storing or not.
- retrievable systems or components.

1.3.2.4. Sepcial equipment.

Descripcion or ealuation of the special systems - and equipments for the documentation, transport - or dismantling of the plant.

1.3.2.5. Social-economic impact.

Study of the influence the closing procedure has in its enviroment, its echosystem and the surrounding population.

1.4. Physical Security

1.4.1. Continuing of the physical protection of the plant, in spite of having the suspension of operation determined.

1.4.2. Physical security programme forecast depending on the chosen decommissioning.

4. CONCLUSIONS

A. As shown, it can be concluded that the decommissioning of a nuclear power plant is an anticipated and necessary activity to be carried out in all nuclear power plants, although the operation time may be variable.

B. The problems, which, pose decommissioning of a nuclear power plant, cause preocupation to those who are working in the nuclear field. The reasons can be very different: economical, technical, hazard surveillance and contro, etc. This variety of circunstances affecting the designers, builders, operators and administration, have a common factor, safety.

C. The decommissioning or closure of a nuclear power plant has a very direct influence on THE NUCLEAR SAFETY INHERENT TO THE FACILITY, THE SITE AND THE FINANCING OF THE PLANT. These three points are the subject of the PSAR, document which the applicant presents as a technical support to the Regulatory Body, that grants permission to the different phases of design, construction, operation and decommissioning of the plant.

D. Finally, if the members of this specialist meeting agree that decommissioning of a nuclear power plant brings along inherents problems of nuclear safety and specific ones about sitting, removal of great components and financing responsabilities, these problems must be dealed before the construction phase. In this case the following could be proposed to the Sub-Committee of Licensing, of which depends the working group that sponsors this meeting.

"TO SUGGEST THE CSNI TO RECOMMEND TO ALL MEMBER COUNTRIES REPRESENTATIVES TO REQUIRE A CHAPTER ABOUT DECOMMISSIONING INCLUDED IN THE SAFETY REPORTS (PSAR, RS, EPS, etc.) THIS CHAPTER SHOULD CONSIDER AT LEAST, THE PROVISIONS OF SITE PROPIETY, THE REMOVAL OF GREAT COMPONENTS AND THE FINANCING RESPONSABILITY THAT CARRIES ALL DECOMMISSIONING PROCESS".

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Second Period of Discussions on Session VIII

K.B. Stadie (NEA) (To M. Perelló)
In assessing the work programme of NEA we have decided to centre activities on decommissioning of NPP under the auspices of the Committee on Radiation Protection and Public Health, recognizing that problems here are primarily on radiation protection and waste management. I have thus some difficulty with your proposal. Nevertheless as you do not suggest a specific programme in decommissioning to be undertaken under CSNI, I propose that you raise your point of considering decommissioning in PSARs, etc., to the SubCommittee on Licensing next week.

M. Perelló (Spain)
I think we should pick up the second part of your comment, as I believe the first is not relevant. The NEA Radiological Protection Committee is more concerned with the methods and means used for decommissioning, and our proposal has to do with the licensing procedure.

P. Giuliani (Italy) (To M. Perelló)
(1) The first is a comment: several countries, like the US, UK, FRG, Spain, do have formal decommissioning provisions.
(2) The question is: do you have information about the 'best' type of NPP as far as decommissioning is concerned?

M. Perelló (Spain)
(1) Certainly, but with a view to permitting the immediate decomissioning. What we propose is that general provision be made before construction. Hence the proposal that the chapter be included in the PSAR.
(2) There can be no such thing, the best method is the one the proprietor is most interested in, unles the Authority makes one mandatory taking into account the forecast in population growth.
As a very complete reference I would suggest you consult Mr. Laurenci's paper presented in Vienna in 1978.

A. Kraut (F.R. of Germany) (To M. Perelló)
Is there any information about cost for decommissioning, especially for dismantling of a heavy water reactor compared with those costs shown for other reactor types?

M. Perelló (Spain)
I have no specific information on the decommissioning of commercial nuclear plants cooled by heavy water, but there is

some on research type reactors such as the Saclay EL-3 reactor.
The most recent information on the cost of decommissioning is in NUREG/CE/0130, which contains a very thorough cost study although it refers only to the decommissioning of a 1000 MWe PWR.

D. Maniori (Italy) (To L. Santomá)
In order to plan and carry out its control activities, does the Spanish Regulatory Body require manufacturers and constructors to submit fabrication and control plans? If yes, what is the policy followed in selecting the plans which shall be submitted? Does the Regulatory Body impose 'hold points' on manufacturing and construction activities, or only 'witness points', or in which other way does it carry out its surveillance duties?

L. Santomá (Spain)
According to the Spanish nuclear regulations, for all the manufacturers of safety related components a fabrication permit has to be granted. Among the documents to be presented when the application is made are the quality control plans and the name of the Independent Inspection Agency, selected by the fabricant and the utility.

P. Lebouleux (France) (To M. Perelló)
Il s'agit plutôt d'un commentaire que d'une question.
Les motifs indiqués pour l'arrêt définitif des centrales nucléaires ne correspondent pas tout à fait à la situation en France. A ce jour, les installations nucléaires ont été arrêtés à des dates correspondant à la durée de vie prévue à la conception (exemple: Chinon I, EL III). Certaines centrales graphite-gaz de type ancien sont maintenues en fonctionnement, soit après des modifications importantes (Chinon III - 450 MWe), soit après modification de leurs conditions limites de fonctionnement et mise en place d'un programme de surveillance plus important.

M. Perelló (Spain)
In effect, there are many research installations and some energy producing ones whose decommissioning has been provided for.
Through my contacts on this subject with Mr. Cregut of the French IPSN, I learned of what CEA has done in such an important area, but I must say that most of the work being done focuses on activities relating to decommissioning methodology and decontamination. Let me refer you to the Proceedings of the Symposium mentioned previously and which was held in Vienna

in November 1978. After much discussion I believe we agreed on the necessity of distinguishing between:
(1) the actions to be taken during the decommissioning; and
(2) the licensing procedure, which should take place immediately before decommissioning, or provision made for it before construction.
As for your mention of Chinon III, in spite of its being commercial it cannot be taken as an example since it comes under the types which, as I said, do not create great problems, as they are within a research or nuclear power centre that will be kept under control because of the presence of other units in the zone.

P. Lebouleux (France)
En fait, je suis d'accord avec M. Perelló, je voulais simplement souligner qu'à ce jour les mises à l'arrêt de réacteur étaient survenues à des époques correspondant aux prévisions. Pour ce qui concerne le programme de mise à l'arrêt auquel a fait allusion M. Perelló, il émane du CEA et ne concerne donc que les installations du CEA.

C. Ringot (France) (To M. Perelló)
En ce qui concerne l'arrêt définitif de centrales, on peut citer un exemple en France où l'arrêt est conditionné par des raisons de sûreté. Il s'agit des réacteurs graphite-gaz G1 et G2 de Marcoule qui ont atteint 25 années d'âge et pour lesquels il a été proposé de poursuivre le fonctionnement.
Les organismes de sûreté ont subordonné la poursuite à l'établissement d'un dossier technique ayant trait aux déformations de l'empilement modérateur. Il a été demandé un suivi permanent de ces déformations, l'arrêt pouvant être demandé si les déformations sont jugées critiques pour un fonctionnement sûr.

M. Perelló (Spain)
Evidently, Mr. Ringot, the public radiological protection argument is a basic one and must be respected whichever method of decommissioning is chosen and whichever reasons lie behind the decommissioning of the installation. From this derives the proposal of having a priori knowledge of the utility's responsibility towards the property and of the Authority's previous requirements for granting it a construction permit.

EXPERIENCES ESPAGNOLES
DANS
L'EVALUATION DES SITES NUCLEAIRES

par

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MME. B. PRESMANES
M. F. RECREO

Resumé

Cette communication présente un analyse des différents facteurs techniques, sociaux et économiques qu'entrent en jeu pour évaluer les sites nucléaires.

D'autre part, elle est presentée pour le cas particulier de l'Espagne, l'organisation de l'Unité d'Evaluation de Sites, et toute la problématique que cette groupe a trouvé pour amener ses fonctions. De différentes solutions sont traitées pour améliorer la situation actuelle.

1. INTRODUCTION

1.1. Exposé général

La problématique que présente l'évaluation des sites d'installations nucléaires est d'une grande ampleur, vu qu'entrent en jeu et se superposent des aspects aussi divers que ceux politiques, économiques et techniques, aspects dont l'imbrication doint tendre à résoudre l'objectif proposé en établissant au préalable, les sources énergétiques appropriées dans les différentes régions, en fonction de leurs recours et de leur économie, faisant particulièrement attention, dans le cas d'une option nucléaire, que la fourniture d'énergie se fasse en toute sûreté pour le public.

D'autre part, le rôle joué par les différents facteurs techniques et économiques et leur degré d'influence dans l'installation même n'est absolument pas homogène, dépendant en même temps des caractéristiques de celle-ci et de celles de son site précis. Le résultat de tout cela se traduit par une grande complexité du problème qui rend coûteux le développement d'une analyse globale satisfaisante recueillant réellement tous les aspects impliqués.

La tentative de donner une solution technique au problème a conduit à établir un ensemble de paramètres en interrelation entre eux, dont l'analyse est faite séparément. De l'ensemble d'analyses partielles doit être obtenue finalement une classification du site étudié. Ce groupe de paramètres établis essentiellement par les principaux pays promoteurs de l'énergie nucléaire et qui ne diffèrent pas basiquement les uns des autres, ont été fréquemment acceptés par les pays comme l'Espagne, dépendants en grande partie des "grands" en matière de technologie nucléaire.

Le problème se pose à nouveau, à l'heure de donner un traitement à l'analyse desdit paramètres, étant donné que les méthodes qui sont employées pour l'évaluation de centrales nucléaires typifiées en trois classes - calcul indépendant, installation de référence et normalisation - considèrent fondamentalement l'analyse de l'installation même et non de son site, car dans ce cas, le concept d'installation de référence est inapplicable dans son sens strict, puisqu'évidemment on ne pourrait jamais parler de site de référence vu l'impossibilité matérielle d'en localiser deux qui pourraient être considérés simplement similaires. L'intégration des deux autre méthodes semble le plus adéquat dans le cas des sites nucléaires.

Compte tenu, comme déjà nous l'avons dit précedemment, de l'hétérogéneité existante entre différents sites, une étude spécifique et détaillée de ceux-ci s'impose; basée dans chaque cas, sur des données fournies directement par les organismes d'évaluation en essayant, ainsi que leurs sources d'information, de ne pas de limiter à la documentation présentée par les entreprises qui sollicitent les différentes autorisations. Parallèlement, les organismes d'évaluation devraient pousser les projets de recherche nécessaires pour supporter les études citées.

Dans le cas concret de l'Espagne et en matière d'évaluation de sites, sont appliqués les méthodes et les techniques des pays d'origine des projets nucléaires, méthodes qui y parfaitement son dûment vérifiées et qui, nous n'hésitons pas à l'affirmer, sont la plûplart du temps et par faute de méthodologie propre, de grande utilité. Il est évident toutefois que pour leur application, il faut disposer d'un ensemble de renseignements basiques sans lequels, très difficilement, peuvent être obtenus des résultats sûrs.

Par ailleurs, la méthodologie utilisée entraîne un ensemble de problèmes additionales d'une part, consubstantiels à celle-ci et liés d'autre part à l'organisation administrative même du pays qui les adopte. Donc, le compartimentage dans l'analyse des paramètres de sûreté qui affectent un site nucléaire, quoique aidant à la meilleure définition de ceux-ci, supporte le problème de la coordination finale de toutes les études effectuées. La grande variation de ces dernières fait que les personnes participant à leur analyse, couvrent une vaste gamme de professions, impliquant souvent des organismes très différents, ce qui fait que les organismes proprement chargés dans chaque pays de l'évaluation d'un site nucléaire, évaluent seulement une partie du problème. En Espagne, jusqu'à présent, l'organisme chargé de cette fonction s'occupe exclusivement des aspects directement liés à la sûreté nucléaire et pas toujours sur son aspect global.

En Espagne et jusqu'à maintenant, on a fait l'évalua tion des sites nucléaires dans les étapes antérieures à la mise en marche de l'installation, semblant important de souligner que le caractère dynamique de ceux-ci avec les conséquentes modifications apportées aux paramètres qui entrent en jeu dans leur définition, étant donné la propre dynamique du milieu dans lequel il s'établissent et dans celle provoquée par la même installation, rend nécessaire une poursuite des études du site au cours de toute la durée de l'installation et même une fois celle-ci déclassée.

Enfin, il est important de mentionner que le grand nombre de problèmes posés dans ces dernières années concernant les site nucléaires en Espagne et l'ambitieux plan nucléaire que l'on prétent effectuer, n'ont pas suffisamment pesé pour élaborer une Planification appropriée qui permettrait de pallier les déficiences existantes, déficiences qui ironten s'aggravant avec le temps, vue que la solution de nombreux principaux problèmes, réclame des périodes de temps no compatibles avec les délais prévus pour l'implantation de nouvelles installations nucléaires.

1.2. <u>Aspects techniques</u>

L'utilisation de l'énergie nucléaire pour la génération d'énergie éléctrique pose, selon ce qui est indiqué précedemment, des problèmes de localisation qui exigent une solution technologique capable de garantir au public potentiellement affecté, qu'il n'y a pas de risques indûs, soit par effets directs du fonctionnement de l'installation, soit en conséquence de la détérioration que tel fonctionnement pourrait suppose au milieu ambiant dans la zone d'influence de l'installation, détérioration qui pourrait introduire des modifications dans l'habitat naturel de l'homme. L'étude du site se révèle donc comme une partie importante dans les études de la sûreté nucléaire.

On peut définir deux types de paramètres dont l'incidence est bien différente: D'une part, il faul considérer les paramètres qui pourraient agir comme des modifiants des systèmes de sûreté et barrières multiples de l'installation de sorte que si l'on n'en tient past compte dans le project, de telles modifications pourraient être la cause du mauvais fonctionnement et qui donneraient lieu à des libérations de radioactivité au milieu. En fonction de la grandeur du terme source, le rang des accidents dont on tient compte dans le projet et qui est considéré dans les rapports d'ambiance, se classe en trois catégories dans chacune desquelles sont définis plusieurs types de faits étant inclus dans le troisième type, les accidents base du projet.

D'autre part, on considère les paramètres du site responsables de la dispersion et de la dilution des effluents

Conformément aux paragraphes précédents, on déduit qu'entre l'installation et le site, il y a une relation biunivoque, de sorte que le couple parfait installation-site est celui dont l'impact dérivé au milieu ambiant est minimum et inférieur aux limites établies par la législation en vigueur. Le problème, du point de vue technique, se centre à sélectionner le couple parfait installation-site. Par conséquent, conception et projet doivent considérer de façon appropriée et avec les coefficients de sûreté convenables, tous les faits naturels ou pas, qui pourraient affecter l'installation. Par ailleurs, le project de l'installation doit englober tous les systèmes de sûreté et de protection capables de réduire le terme source produit dans des circonstances accidentelles ou de fonctionnement normal, à des valeurs acceptables, introduction qui devra tenir compte des caractéristiques de diffusion et de dispersion des milieux récepteurs.

Tout ce qui précède est conforme à la philosophie générale de sûreté nucléaire, basée sur les concepts de "Defense in Depth" et de "Design basis envelope", interpelle directement tous les paramètres du site qui pourraient amener à une faille dans l'installation, provoquant des libérations incontrôlées de radioactivité. En conséquence, on doit pouvoir établir que la probabilité composée définie par la conjonction d'un événement extérieur et par la probabilité de que cet événement represente une faille dans l'installation, soit inférieure à la limite de "Design basis envelope" dont la valeur est située aux environs de 10^{-6}.

Les principes que nous venons d'exposer conduisent à l'établissement de critères qualitatifs de sûreté des sites, critères qui définissent le problème, mais ne le quantifient pas.

Le besoin de quantification dans le processus d'évaluation des sites mène à introduire le concept de risque, défini comme la potentialité d'un dommage nucléaire. Pour fixer le maximum de risque admissible, il y a deux types d'approximations à l'équation fondamentale. Le critère mécaniciste considère le dessin de l'installation sous l'hypothèse que va se produire le maximum d'accident prévisible, ce qui ne représente pas une évaluation réelle du risque. Le critère probabilistique estime le risque en le définissant comme l'éspérance mathématique d'un dommage nucléaire, compte tenu de tous les faits possibles. Nous comprenons que l'application de la méthode probabiliste exige une séquence méthodologique qui implique deux pas essentiels:

(a) Déterminer la probabilité et la grandeur du dégagement, et

(b) Déterminer les conséquences dérivées au public.

En suivant cette séquence méthodologique pour évaluer un site nucléaire, l'ensemble de paramètres le déffinissant se scinde en deux groupes. D'une part, les paramètres qui ont de l'influence sur l'établissement de la probabilité que se produise un dégagement en conséquence d'un fait externe et, d'autre part, les paramètres qui déterminent les conséquences découlant des dégagements radioactifs.

L'ensemble de paramètres du site ayant de l'influence sur l'établissement de la probabilité de libération, sont les paramètres géologique, sismologique, hydrologique et météorologique.

Les phénomènes géologiques qui ont une incidence dans le domaine de la sûreté nucléaire doivent être définis en fonction de leur cinétique ou à partir d'un nombre suffisant de données permettant de traduire la fréquence d'une observation déterminée au concept de période de retour. La dynamique des phénomènes géologiques exige de considérer de façon préférentielle le paramètre structural, lequel du point de vue de la sûreté nucléaire, reste défini dans le concept de faille active et des caractéristiques néotectoniques de la zone.

Les pheonomènes sismiques doivent être analysés selon deux points de vue: fréquence de séismes enregistrés dans la région et caractéristiques de cette sismicité. L'analyse de la fréquence des séismes intéresse sur les aspects concernant la sismicité historique, quoique celle-ci ne fournit aucun genre d'information sur les caractéristiques de l'origine du phénomène sismique, aspect d'importance essentielle et basique pour le projet antisismique des structures puisque les paramètres qui définissent le mouvement du sol (input dans le calcul de réponse de la structure) sont fonction du mécanisme focal et du milieu de propagation. D'où la convenance que dans notre pays s'installe un vaste réseau d'observation instrumentale, afin que puisse être entreprise la réalisation d'une carte sismotectonique appuyée sur des données tectoniques et sismiques instrumentales. Cela permettrait d'appliquer des spectres de réponse qui seraient conformes au caractéristiques du site problème. L'application du spectre de réponse recommande dans la guide régulatrice 1.60, pourrait ne pas être applicable à quelques-uns des sites espagnols, car celui-ci est établi en considérant la moyenne plus une déviation typique de plusieurs spectres correspondant à des phénomènes sismiques de type californien.

Les études hydrologiques pour déterminer la cote de nivellement de l'installation, doivent se baser sur la détermination des débits maximums des crues à espérer sur le site, déterminations qui exigent de considérer l'histoire géologique de la région en fonction des caractéristiques climatologiques, géomorphologiques et hydrogéologiques du bassin hydrologique affecté. Cela représente l'application de modèles basés sur les valeurs extrêmes pour déterminer la probabilité d'engendrer la débit maximum basé sur une période de retour définie au préalable.

Les caractéristiques météorologiques qui ont de l'influence sur la sécurité de l'intégrité structurale des bâtiments directement ou indirectement en rapport avec la sûreté nucléaire, sont les valeurs des charges maximums dûes au vent ou aux précipitations. Pour calculer les charges maximums, il faut étudier la fonction de distribution de probabilité définie en fonction du temps pour les valeurs du vent - vitesse et direction - et des précipitations.

La correcte évaluation des paramètres antérieurs fournit les données d'entrée nécessaires au projet, afin que puisse être garantie l'integrité des structures sur les apsects qui font référence aux faits externes naturels.

La seconde partie d'un processus d'évaluation du site nucléaire, surgit de l'hypothèse de l'occurrence d'un fait qui pourrait donner lieu à de libérations de radioactivité et l'objective se centre, après avoir analysé les caractéristiques de dispersion et de dilution des milieux primaires de transfert, à déterminer l'incidence que ces dégagements radioactifs produisent chez l'homme. Pour fair cette étude, il faut connaître les caractéristiques du milieu physique affecté et celles du déversement, à partir desquelles on procède à établir les modèles de diffusion atmosphérique appropriés pour des déversements gazeux et ceux de dispersion hydraulique pour des déversements liquides. Comme en définitive, cette partie de l'évaluation centre son objectif sur la protection du public qui pourrait se voir affecté aussi bien par le fonctionnement normal de l'installation qu en cas d'accident, on étudie dans cette phase la distribution de la population dans la zone d'influence, selon les prévisions à la date de mise en marche et déclassement de l'installation étudiée.

Le calcul du risque se fait à partir des résultats des deux parties de l'étude, comme un produit de la probabilité de que se produisse un mauvais fonctionnement dû à un fait externe par le dommage ou les conséquences découlant de celui-ci.

Conformément à ce qui est indiqué précédemment, nous comprenons que l'évaluation d'un site nucléaire constitue une tâche multidisciplinaire dans laquelle le résultat excellent ne s'obtiendra pas seulement par optimisation des parties, mais sera fonction en même temps de l'imbrication appropriée qui est donnée à celles-ci; d'où l'importance de donner à un groupe de cette nature, l'organisation adéquate, suffisamment souple et rationalisée pour atteindre l'objective proposé. Ce thème sera développé dans les paragraphes suivants, et sur la base de l'expérience acquise pendant les années où le Groupe des sites a dévelopé son travail au sein de l'Unité Opérative d'Evaluation du Département de Sûreté Nucléaire de notre JEN.

2. ORGANISATION ET METHODOLOGIE DE TRAVAIL DANS L'EVALUATION DE SITES NUCLEAIRES EN ESPAGNE

2.1. Situation Actuelle

Depuis la création du Groupe d'Evaluation des sites (mars, 1974), date qui représente la reconnaissance officielle de la part du Service de Sûreté Nucléaire de la problèmatique spécifique du site des installations nucléaires et radioactives, les critères ont évolué, en fonction de la situation que les changements politiques et sociaux ont configurée dès lors.

En particulier, le critère initial qui a donné origine au Groupe, a été l'évaluation de la documentation relative au site, liée essentiellement à la phase d'Autorisation Préalable. Jadis, la structure du Département de Sûreté Nucléaire était composée par trois Unités Opératives - U.O. Evaluation, U.O. Protection, U.O. Inspection - et le Groupe d'Evaluation des sites s'est assigné à la U.O. d'Evaluation, de sorte qu'au début la fonction fixée à chaque évaluateur était multiple en ce qui concerne les disciplines et unique quant au Projet assigné. Cela signifiait qu'un évaluateur de formation spécifique dans un des paramètres du site, tel qu'on le décrit dans l'épigraphe 1.2, aurait à traiter sur tous les paramètres d'un même site. Rapidement, on a remarqué la double nuance négative que tel procédé offrait; d'un côté, la difficulté qu'une personne traite de thèmes si dissemblables comme peuvent l'être la géologie et la démographie, et de l'autre, la présence d'un seul critère devant un problème, non seulement multidisciplinaire, mais de la transcendence d'un site nucléaire.

Dans l'intention de surmonter la situation antérieure, on a essayé d'assigner aux évaluateurs, des thèmes se rapprochant de leurs respectives formations basiques: de cette façon, et vues les caractéristiques des personnes qui, en ce moment, faisaient partie du Groupe d'Evaluation des sites, les thèmes se rapportant à la Géologie - dans le sens large-, à l'Hydrologie et à la Météorologie, sont restés couverts. Grâce à ce critère, on a réussi, au moins, que dans chaque projet coexistent plusieurs critères individuels et que chaque évaluateur traite des thèmes plus en rapport avec leur formation basique, ce qui ne détériore pas les critères globaux que doit posséder un évaluateur, mais qui sont plus puissants et se développent en travaillant en équipe. Devant ce nouveau tracé, loin toutefois de laisser résolue la problèmatique d'évaluer un site nucléaire, de nouveaux problèmes se rapportant également, comme nous le verrons, aux aspect d'organisation, sont mis en évidence.

Si nous prenons comme valable la séquence méthodologique exposée dans l'épigraphe 1.2., nous remarquons que dans "l'analyse des conséquences" elle doit partir des données hydrologiques du milieu pour ultérieurement, calculer l'impact dérivé à l'homme en conséquence des déversements radioactifs. Sur cet aspect, on a mis en évidence quelques deficiences qui réduisent l'efficacité puisqu'il n'y a pas eu une bonne et directe liaison entre le Groupe des sites qui doit fournir les données d'entrée précédemment citées, et l'Unité Opérative de Protection qui doit déterminer les conséquences découlant d'une libération de radioactivité pour des conditions normales ou en cas d'accident.

D'autre part, il existe également des lagunes qui se traduisent par une faute d'utilisation du travail de certains secteurs. Par example, les résultat obtenus dans les analyses géologiques et sismologiques devraient être utilisés para le groupe de génie civil, groupe qui n'existe pas au moment où sont écrites ces lignes.

Toutes ces anomalies de fonctionnement ont été recueillies, à titre d'expérience accumulée, profitant du moment conjonctural que vit notre pays en matière d'énergie, et seront soulignées en cas de consultation en ce qui concerne l'Organisation du Counseil de Sûreté Nucléaire, dont la Loi de Création se trouve en ce moment au Parlement Espagnol, en attente de débat.

2.2. Considérations vis-à-vis de l'avenir

L'incapacité de l'organisation sur laquelle repose actuellement l'évaluation des sites de donner une réponse satisfaisante aux questions que la société espagnole et en particulier, les communautés les plus affectées se posent concernant le niveau de risque associé aux sites sollicités, a mis en évidence l'impropre de celle-ci. Le critère initial qui a donné origine à cette Organisation - tel qu'on l'a indiqué précédemment - a été dépassé par la dynamique sociale espagnole de plus en plus consciente de la singulière importance que la situation des installations nucléaires et radioactives atteint dans l'aménagement territorial. En conséquence, l'objectif générique de l'organisation que nous prétendons, ne pourra pas être circonscrit à l'étroite marge que l'Autorisation Préalable définit dans la Règlementation nucléaire espagnole, vue que l'évaluation d'un site pour une installation nucléaire ou radioactive de première catégorie, excède de beaucoup le caractère préliminaire des données qui peuvent appuyer la solicitude de cette autorisation, si nous nous en tenons à la Règlementation mentionnée. D'autre part, l'évaluation d'un site nucléaire ne peut se passer de la connaissance détaillée des structures qu'il devra héberger, ni, non plus, d'une estimation sûre des effluents auxquels l'installation prévue donnera origine, les deux aspect étroitement liés au projet de celle-ci. Il reste ainsi souligné que la portée qui a été donnée à la définition de l'Autorisation Préalable quant à "reconnaissance officielle du site choisi", au point de considérer telle autorisation comme synonyme d'autorisation "du site", ne doit pas durer plus longtemps, du moins si l'on prétend éviter les erreurs indéniables auxquelles a mené son application.

Un autre des aspects qui doivent être considérés obligatoirement en projetant une organisation opérative, est celui des moyens dont on dispose. Nous distinguerons, formellement, entre moyens humains et moyens matériaux, bien que leur complémentarité résulte aussi évidente que leur non permutabilité.

Il sera certainement discutable si un Organisme de Sûreté Nucléaire devrait contenir en lui même, toute la capacité technique nécessaire pour couvrir la variée casuistique pouvant être object d'évaluation de risques. Ce doute est possible également quant à la sélection et à l'évaluation de sites pour les installations nucléaires et radioactives, surtout en considérant que les disciplines involucrées sont, pour la plupart, de caraetère conventionnel et objet, par conséquent, d'autres Organismes officiels, dans quelques cas avec caractère de concurrence exclusive. Il semble approprié, néanmoins, pour l'Organisme de Sûreté Nucléarie, de parvenir à l'exclusivité de l'application de telles disciplines à l'analyse de sûreté. Tel est le cas de l'évaluation des sites nucléaires, sans que cela puisse représenter une intromission dans les domaines d'activité d'autres organismes. D'autant que l'Organisme de Sûreté Nucléaire devrait garder des relations statutairement reglementées avec ces organismes, afin de profiter de l'expérience de ceux-ci et de se rendre plus légère la charge du nombre élevé de techniciens dans un autre cas nécessaires, avec le risque de transgresser des compétences étrangères. Or, une collaboration ne peut représenter, sinon tout le contraire, la possibilité de méconnaissance par l'une des parties des matières objet de collaboration. L'organisation chargée de l'analyse des sites est donc obligée de disposer du nombre adéquat de techniciens permettant d'affronter les problèmes qui pourraient se présenter dans l'évaluation de risques et de donner à ceux-ci la formation spécifique que réclamerait l'évolution des disciplines correspondantes.

Actuellement, le Groupe d'Evaluation de sites maintient des collaborations pour l'élaboration de ses rapports avec l'Institut Météolorologique National et avec l'Institut Géographique National. Evidemment, sont beacoup plus les Organismes officiels auxquels on devrait demander de collaborer à un sujet qui dépasse largement la capacité de l'organisation actuelle. Une telle collaboration, qui en rien ne peut diminuer l'irrenonçable indépendance du Organisme de Sûreté Nucléaire, non seulement contribuerait à une meilleure connaissance des problèmes et à une plus grande capacité d'analyses, mais éviterait, en outre, la paradoxale situation de devoir faire un rapport sur des études qui, présentées par le sollicitant, on été faites par l'un de ces organismes.

Avec les Communautés Autonomes les matières spécifiques de collaboration le seront quant à l'aménagement du territoire et en matière de protection du milieu ambiant, selon les compétences reconnues à celles dans la Constitution. Cette collaboration ne pourra par conséquent, se limiter à la phase de sélection de sites, mais devra se prolonger durant l'opération de l'installation. Ainsi pourraient être palliés les raisonnables soupçons que la planification actuelle de sites nucléaires fait apparaître dans les régions les moins developpées et qui en Espagne, sont souvent, en outre, fortement excédentaires en énergie électrique produite.

Quant à la disponibilité des moyens matériaux (essentiellement instrumentaux et de documentation), quoique de tels moyens ne peuvent substituer ni remplacer les humains et ils serviraient bien peu sans une adéquate dotation, redondance et actualisation des techniciens chargés des évaluations, leur inexistence ou leur précarité conduirait à des résultats médiocres.

Bien que la méthodologie de l'évaluation d'une installation nucléaire ou radiactive doit considérer l'étude détaillée des caractéristiques technologiques de celle-ci, dans ce qui suit et aux effets de proposer une organisation pour l'évaluation des sites, nous tiendrons seulement compte, pour ce qui concerne celle-ci, des problèmes découlant des interphases installation-site, en particulier ce qui se rapporte aux paramètres basiques de project d'une part et ce qui concerne les émissions de celle-ci aussi bien en opération normale qu'en cas d'accident, ainsi que concernant les systèmes de traitement des correspondants effluents liquides et gazeux, de l'autre. L'organisation que nous contemplons pourrait effectuer les études auxquelles donneraient lieu les processus d'évaluation des sites de deux façons: soit en les réalisant de nouveau, indépendamment de l'entité qui les soumettrait à consideration - généralement l'entreprise exploitante - soit en révisant et en analysant les études déjà faites par le sollicitant, gardant dans les deux cas la faculté d'exiger à celui-ci les études complémentaires qui seraient considérées opportunes. Le choix de l'une ou de l'autre voie introduit de grandes différences dans la composition, l'organisation, la méthodologie et les moyens - humains et économiques - de l'organisme régulateur. Assurément, la capacité de fair par soi-même les études exigibles aux sollicitants confère à l'organisme régulateur le plus haut degré de solvabilité. La complexité des études requises, le coût associé et l'existence d'autres organismes officiels concurrents en partie de ceux-ci, permettent de douter que ce système soit le plus approprié pour un pays de la capacité économique de l'Espagne. De toute façon, même si l'organisation doit être capable, techniquement et économiquement, de fair les études qu'à son tour elle exigerait, comme si seulement elle devait jouer le rôle de supervision, nous comprenons qu'elle doit respecter les principes de compétence, d'indépendance et de redondance comme base irrenonçable de leur définition.

Par <u>principe de compétence</u> s'entend ici celui que les analyses des différents paramètres soient faites par du personnel objectivement qualifié, de telle sorte que ne se produise pas de perte de capacité entre les members de l'Administration chargés de l'analyse par rapport aux auteurs des études correspondantes. Malgré l'évidence de ce principe, il est indéniable qu'il n'a pas été respecté dans tous le cas. Le principe d'indépendance a deux aspects. D'une part, indépendance des organes techniques des intérêts des promoteurs du projet qui sont sous l'analyse; indépendance économique, administrative et juridique. D'autre part, indépendance quant aux sources de renseignements, ce qui représente une liberté de recherche, de consultation, d'expression et le pouvoir de ne pas révéler les sources d'information. Le troisième principe basique est celui de redondance. Un organisme régulateur en matière de sécurité nucléaire doit, comme but à assurer, réduire la possibilité d'erreur humaine au minimum pouvant se produire. D'où le principe de redondance. Redondance dans les sources de renseignements, évitant de tomber dans l'analyse ou la considération de sources d'une même provenance; redondance dans le méthodes d'analyse, pour éviter ainsi l'erreur méthodologique, mettre en évidence les limitations de celles-ci et marquer les erreurs inhérents aux données; redondance de critères, englobant ce qui précède et qui l'amplifie en ce qui concerne la définition et la formulation d'hypothèses, aspect ce dernier qui revêt un grand intérêt dans l'analyse d'hypothèses de faible probabilité associée. Enfin, et de façon notable, redondance dans le personnel technique chargé de faire les analyses, de sorte que les techniciens y participant soient, simultanément et exclusivement, dans l'étude d'un même paramètre dans un même projet. Les trois principes indiqués comme basiques soutiennent la confiance témoignée à l'organisation chargée de l'évaluation, aussi bien dans le cas de l'analyse de sites que de tout autre système de l'installation, mais qui dans le cas concret de l'évaluation des sites revêt un plus grant intérêt, sans doute, par l'inapplicabilité de la méthode de référence, les limitations de la normative et la complexité de l'application des méthodes de calcul.

Après avoir exposé les aspects administratifs où se situe l'intervention d'un organisme de sûreté et les principes philosophiques basiques qui doivent régir cette intervention, il semble convenable de reconsidérer les aspects techniques développés dans le paragraph 1.2. de cette communication, afin de structurer l'organisation d'une manière présomptive la plus appropriée pour l'évaluation des sites dans un pays de la dimension économique de l'Espagne. Nous y distinguions deux sortes de paramètres du site: ceux qui peuvent amener à des charges sur l'installation et ceux qui définnisent les milieux primaires de transfert. De tels paramètres ont une incidence de façon distincte au cours du processus d'évaluation, depuis la phase de sélection du site jusqu'au déclassement de l'installation. En conséquence, les différentes phases d'autorisation devront souligner les différentes fonctions de chacun des paramètres analysés.

En sollicitant l'autorisation préalable, on devra pouvoir établir les valeurs des paramètres du site qui interviennent dans la définition de "Design basis envelope" et la compabilité de celui-ci avec l'impact radiologique, chimique et thermique dû a l'ensemble des conditions les plus défavorables en régime d'opération normale et dans le cas de l'accident maximum prévisible. Précédemment au permis de construction, devraient être parfaitement définies les conditions géotechniques du site, la disposition des bâtiments de l'installation, les systèmes de refroidissement de l'usine, dans le cas des centrales nucléaires et les caractéristiques de dilution et de dispersion atmosphérique et aquatique. La richesse des données spécifiques du site dans cette phase obligera à un puissant traitement numérique inabordable en marge du calcul automatique. Par ailleurs, les travaux sur le site tranchées, condages, études géotechniques et géophysiques et principalement, les fouilles préalables à la fondation, doivent être contrôlés au moment de leur réalisation, non en fonction des délais ou des phases administratives d'autorisation, comme c'est le cas actuellement. Aussi bien le contrôle "in situ" des travaux effectués sur site qui pourraient affecter la sûreté des installations que la révision du projet, représentent une infrastructure et une dotation de moyens bien supérieurs à ceux disponibles pour le moment, où les rares contrôles sur chantier ne peuvent aller au-delà de pures inspections visuelles.

La phase d'opération doit permettre de perfectionner les modèles de dispersion, de dilution et de transport des effluents radioactifs dans les milieux primaires; d'évaluer l'influence des paramètres d'ambiance, comme un tout, sur les ecosystèmes et de calculer donc, les doses réelles à la population affectée à courts termes (estimations mensuelle ou trimestrielles) et les prévisions à moyen terme sur les bases des données réelles obtenues. A nouveau, la complexité des modèles applicables et la diversité et nombre élevé des données requises obligent à disposer d'un support informatique adéquat et d'une organisation dotée pour l'obtention et l'exploitation de ces données, indépendamment du titulaire de l'installation.

L'organisation chargée de l'analyse des sites nucléaires devra tenir compte des besoins mentionnés. L'effort à fournir, etant donné la diversité des tâches imbriquées dans le processus d'évaluation, la multiplicité et le volume des renseignements nécessaires, leur obtention, dépuration et traitement, demandera certainement, le concours d'autres organismes de l'Administration. Dans ce qui suit, nous indiquerons le schéma d'organigramme qui est jugé approprié pour l'évaluation des sites et les organismes de ceux dont il semble nécessaire d'obtenir la collaboration. Tout d'abord, il semble conseillable que cette organisation, placée au niveau administratif qu'il conviendrait, dépende hiérarchiquement du maximum responsable technique de l'Organisme Régulateur en matière de Sûreté Nucléaire par un coordonnateur général et directeur technique de l'Unité d'Evaluation des sites, responsable unique de celle-ci devant lui. L'unité devrait posséder une indépendance technique et économique dans le cadre de sa compétence. Les domaines d'intervention à couvrir par cette Organisation comprendraient l'analyse de sûreté du site et l'analyse de l'impact d'ambiance. A cet effet, l'Unité devrait être structurée dans les sous-unités de Paramètres Physiques chargées de l'analyse des paramètres physiques de l'emplacement ou facteurs techniques sitologiques (géologiques et de transfert) et de la définition des paramètres base du projet; la sous-unité des Etudes d'Ambiance,pour l'étude des paramètres biologiques, facteurs démographiques, bioécologiques, sociaux et culturels, et la sous-unité d'Evaluation de l'Impact d'Ambiance, chargée du traitement des effluents, de l'évaluation de l'impact radiologique dans des conditions d'opération normale et en cas d'accident et de l'analyse du coût/bénéfice.

Pour le meilleur accomplissement de ses buts, l'unité d'Evaluation de sites devrait compter sur les correspondants services techniques spécifiques, unités mobiles pour enregistrement et prise des données, laboratoires d'analyses (pétrophysique, chimique, radiochimique et biochimique), unités de calcul, archives et documentation, en outre du support administratif adéquat.

Cette Organisation permet de conjuguer la diversité des tâches en rapport à l'évaluation des sites et l'économie nécessaire des moyens, dans un pays aux caractéristiques de l'Espagne. Les fonctions restent couvertes ainsi en extension, mais pas en profondeur, étant donné leur complexité. Il faut par conséquent d'autres Organismes de l'Administration en sollicitude de conseils techniques ou de documentation. Sans que la relation puisse être considérée complète, en plus de collaborations actuelles avec l'Institut Météorologique National et avec l'Institut Géographique National, il est conseillable de les initier avec l'Institut Hydrographique de la Marine et l'Institut et Observatoire de la Marine, du Ministère de la Défense; avec l'Institut d'Etudes d'Administration Locale, du Ministère de l'Intérieur; avec le Centre d'Etudes et d'Expérimentation des Travaux Publics et les Confédérations Hydrographiques, du Ministère des Travaux Publics et Urbanisme; avec le Conseil Supérieur de Recherches Scientifiques, du Ministère des Universités et de la Recherche; avec l'Institut Géologique d'Espagne, du Ministère de l'Industrie; avec l'Institut National de Réforme et de Développement Agraire et avec l'Institut National pour la Conservation de la Nature, du Ministère de l'Agriculture; et avec l'Institut Espagnol d'Océanographie, du Ministère des Transports et des Communications.

3. FORMATION D'EXPERTS COMME ELEMENT FONDAMENTAL DANS LA GARANTIE DE QUALITE D'UNE EVALUATION

Le problème de l'évaluation des sites nucléaires se révèle, à partir de ce qui est exposé antérieurement, comme un problème difficile à aborder par sa complexité, et le degré de la fiabilité des résultats obtenus dans un processus d'évaluation, sera fonction aussi bien des schémas d'organisation que de la qualité technique des évaluateurs, les deux aspects ayant une incidence mutuelle.

La qualité technique d'un rapport d'évaluation dépendra, en plus de moyens physiques et humains donnés, du critère même de l'évaluateur sur le thème soumis à jugement. Il faut tenir compte que le niveau de critère minimum exigible au cas, sera atteint non seulement en fonction du temps et du nombre de travaux réalisés par celui-ci, mais également et surtout, par le type et la nature de l'entraînement fourni au future évaluateur.

Pour ce qui est de l'Organisation, il est essentiel, en dehors de ce qui est indiqué dans l'épigraphe précédent, que l'évaluateur dispose d'une liberté totale pour faire toute sorte de consultations à des centres de recherche, universités, etc.

L'assistance à des Cours et à des séminaires se révèle positive quant à la mise à jour et l'apprentissage de nouvelles techniques, mais l'introduction des évaluateurs à des Projets de Recherche comme partie active de ceux-ci peut être très utile - dans le processus de formation de "critère" à quoi nous faisions mention ci-dessus.

Nous espérons que toutes les considérations notées dans le présent travail pourront être discutées avec les représentants d'autres pays, car avec leur expérience, ils peuvent nous indiquer possibles chemins à suivre devant l'expectative à laquelle est soumise la Sûreté Nucléaire dans notre pays et au moment actuel.

CSNI SPECIALIST MEETING ON REGULATORY

REVIEW IN THE LICENSING PROCESS

Madrid, 7th-9th November 1979

AN EVALUATION MODEL FOR THE DEFINITION OF REGULATORY REQUIREMENTS ON SPENT FUEL POOL COOLING SYSTEMS

ABSTRACT

An evaluation model is presented for establishing regulatory requirements in the spent fuel pool cooling system. The mayor design factors, regulatory and design limits and key parameters are discussed. A regulatory position for internal use is proposed. An associated numerical model has been developed and its salient features are shown. A simple formula for calculating temperature delays and maximum temperatures under normal conditions is presented. Finally associated problems and experience are discussed.

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AN EVALUATION MODEL FOR THE DEFINITION OF REGULATORY REQUIREMENTS ON SPENT FUEL POOL COOLING SYSTEMS

I. Introduction

a) Spain Licensing Process Aspects.

Within the frame of the Spanish Licensing Review and evaluation process both new and old facilities necessitate at - this time an evaluation of the spent fuel pool cooling system, SFPCS.

For new plants, the issue of the "Provisional Operating License"[1] (permiso de explotación provisional, P.E.D) require it, and if spent fuel racks will be used for early fresh fuel storage, it seems additionally reasonable to perform the evaluation before granting the "Fuel Storage Permission", also required in our regulations[1].

For old plants, world reprocessing policies provided strong incentive for increase of storage capabilities. Changes in regulations since the time of startup made it convenient to implement the new standarts before allowing the facilities to continue operation with more spent fuel stored. This implementation is also consistent with the philosophy for the issue of the "Definitive Operating License"[1] (permiso de explotación definitivo, P.E.D). In some cases, fuel transportation and reprocessing constraints pushed this subject with urgency.

b) Diversity of installations. Independent facilities.

Although the basic design principles are the same, there is considerable scattering in the way of implementing them in the whole spectrum of old-to-new facilities. On the other hand, prospects for future construction of an independent storage installation, additionally diversify the technical solution of the cooling problems. Such diversity necessitates practical evaluation guidelines beyond those included in the regulations comented below. In particular, the demostration of quantitative criteria that usually determine the software requirements on the system, i.e. its operational limitations, ougth to be carefully defined and applied.

Urgency, unified treatment and unambiguous interpretation of cuantitative aspects of the regulations, were then claims for the development of a cuantitative licensing evaluation model.

II. Mayor Design Factors. Interfases and Operational Implications.

The areas of concern are interrelated through the following design considerations:

a) Main process variables.

They are the allowable average and outlet pool temperatures, T_{avg} , T_{out} , and allowable pool levels under normal, abnormal and accident situations. The regulatory limits in these process variables place funtional requirements on the SFPCS components and are the logic flow starting points of licensing action.

All of them*, however, will be based upon maintenance of the general radiation and exposure limits of the radiological protection regulations (10CFR20-100 and Appendix I to 10CFR50 in the U.S.A. regulations[2] context). Note that fuel temperatures are not included, because they impact the rack design rather than the spent fuel pool cooling system. Interfases between both should anyway be assessed.

b) Relevant Items.

The limits before, will force attention to the following items:

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* Process variable limits originate the "barrier protection" while radiation and exposure limits originate the "radiological protection" as defined in reference 3.

i) Limit the maximum decay heat loads (peak design type).

ii) Ensure reliability of the cooling system components

iii) Ensure protective measures

iv) Ensure initial conditions of transients

v) Ensure power supply, the intermediate cooling (like component cooling system) operation and the ultimate heat sink

c) Design parameters and characteristics associated to each item.

They include:

i) Cooling times before fuel discharge, t_Δ , time to discharge, t_o, number of fuel elements, N_i, per discharged batch i and its operational history, $P_i(t)$, as planned in advance in fuel management schemes, with irradiation time T_i, and cooling time τ_i

ii) Number and arrangement of spent fuel cooling pumps and heat exchangers, pipe arrangements and flow derivations to other related systems (cleanup), backup and protective systems connections, primary and secondary water source reliability as well as its associated software like primary, G, and secondary, G_s, heat exchanger flows, and coefficients, U, and surfaces, A, of exchangers heat transmission.

iii) Backup and redundant cooling and make-up systems, its associated software including times to perform required connections (for non-permanently installed systems), make-up system capacities and the availability of those systems for spent fuel cooling purposes; component safety classification; instrumentation for pool level and temperature measurements and its safety cualification, including detector locations and measure mecanisms.

iv) In-service inspection of pipes, frecuency of temperature readings, control room indications, minimum and maximum cooling flow maintained and standby systems maintenance.

v) Time to start emergency power supply under loss of off-site power conditions, water sources and mecanisms for assuring a maximum value of secondary side water temperature and cooling flow, including capacity of this system for SFPCS purposes.

d) Operational implications.

i) Possible delays in reloads or constraints in reload operations

ii) Monitoring and instrumentation requirements

iii) Need for standby systems maintenance

iv) Power supply requirements

v) Some limitations on reload schedules, amount of fuel discharged and fuel management schemes.

vi) Constraints on operations with defective fuel

vii) Lack of confortability of pool operations

III Regulatory Requirements

Because most of the spanish plants are NSSS-U.S.A. technology, coherence of the integral design recommends the use of the american standarts as a reference for any internal position. This section is then devoted to a) review different U.S.A. positions, in as far as we know them, both from industry and regulatory sources b) show the difficulties of clarifying differences and potential contradictions in particular aspects, and c) propose a position for internal use within the JEN Nuclear Safety Department.

a) Review of U.S.A regulatory and industry safety barrier criteria.

In this paper we will only deal with the second barrier, i.e. the pool water, because it is the relevant one for SFPCS design. Emphasis will be in process variable quantitative limitations under different classes of events, limitations considered as sufficient second barrier safety criteria.Without a clear-cut position on any cuantitative limit, detailed calculations for demonstrating compliance may be meaningless.

Table I shows design limits from the most commonly quoted regulatory and industry sources. Both, event classification and requirements are different and there are also differences within closely related sources. Important questions like whether or not the SFPCS should be seismically designed are differently answered. Also requirements for alternate cooling availability remain unclear. The actual application of the different positions is hard to track and we have found examples of all of them without apparent reasons for its selection. We do consider such a situation just a simple, not very important, example of the difficulties of "importing regulations" in parallel with imported technologies. Note that no european regulations have been added to the list, a fact that will however become necessary with other foreign plants.

b) Implications.

As mentioned in ref. 3 some kind of logic flow is required in these cases for finding an useful approach to assimilate the experience contained in the available standarts, minimizing the risk of false interpretations. The importance of event classification and associated exposure and radiation limits was there[3] emphasized and has been reflected for this particular case in table I.

To show an example, consider an event like the rupture of the cooling system inlet pipe within the pool (Fig. 1). Such an event may be classified[*] as condition III within the ANSI-210[4] - terminology (actually pipe and tank leakages are mentioned in the examples given in the guide), but should be considered an abnormal event within the GDC classification. Exposure and radiation limits are then 10CFR100 and 10CFR20 respectively. The "minimum shielding water depth" design requirement should then be also different, although this is unclarified in ANSI-210[**]. As a result

* Note that the rupture may be inadvertent for long times unless appropiate in-service inspection is considered.

** A possible interpretation is to maintain "emergency condition limits of 10CFR20" rather than "normal condition limits" suitable for condition II.

pool boiling could be allowed in the first case but not in the second. Because this event introduce large axial temperature gradients, T_{avg} and T_{out} are different, the concept of level may be jeopardized and both the measured level and temperatures will depend on the type of measurement. To comply with GDC criteria, additional constraints on initial pool temperatures, pipe in-service inspeccion, temperature detectors alarm set point and temperature and level measure-mecanism requirements may be needed, constraints that will not pressumably be necessary to demostrate compliance - with ANSI-210. Similar things may be said about the "loss of non-category I portions of the SFPCS" also classified as condition III in ANSI-210 which involve requirements on alternate seismic cooling systems.

c) Proposed position.

Our proposed position was adopted with a view to maintain the GDC classification and associated requirements and to include interesting features incorporated in the more modern ANSI-210 in those aspects where different regulatory interpretations have been observed. The position is shown in table II.

We have distinguished between T_{avg} and T_{out} limits because essential equipment like pool walls concrete (T_{avg}) and cleanup system resins (T_{out}) should work within its design limits. The 150ºF, 140ºF figures are used for both purposes in ANSI-210 and - the Standart Review Plan[5] (SRP), although the second limit usually implies the maintenance of the first. The 125ºF limitation is not a requirement, as mentioned in ref. 6, but a "nice to have" for confort reasons. In the case of transients, it is important to take into account the difference between T_{avg} and T_{out}, especially when the latter is the only one measured and alarmed.

The requirement "not to allow boiling under condition II events with corrective measures single-active-failure proven" serves to clarify the degree of alternate cooling and redundancy of the SFPCS that may be needed, as well as the acceptability of the required connection operations.

As an example, consider two arrangements with one operating pump, A, and a redundant pump, B, respectively "on line" or "manually connected" and assume the anticipated operational occurrence resulting from a failure in the"A"pump, under normal heat load conditions. In the latter, the operator need to "start" pump B; operation that will require a certain time. Now, a failure of this corrective measure should be assumed and a third alternate cooling means should prevent boiling. The same but shorter sequence will be followed in the "on-line" system, reducing its alternate cooling capacity needs.

Assume now the on-line system under maximum heat load conditions with both pumps operating and the same event. Because the other pump continues to run, is not a corrective measure, need not to be failed and may be enough to prevent boiling. This is not the case of the other system, that should be subjected to the same sequence before, now with maximum heat load conditions, increasing much more the requirements on alternate cooling.

In the same way, the reliability requirement of SRP relative to loss of offsite-power, is also clarified with this criterion, provided loss of off-site power is classified as an anticipated event (example of condition II event mentioned in ANSI-210).

Finally, in the case of accidents where site evacuation is assumed, alternate cooling is not required (10CFR20 is not pertinent) but connection of the make-up system should be made before boiling (with allowance for additional single failure) and the make-up system capacity will be judged requiring the fuel to remain covered assuming the replacement of all of the evaporated water. Usually, off-site dosis are not a problem from a radiological - stand point.

IV Evaluation Model

a) Basic Equations.

The model has been developed to account for all of the event conditions (normal, anticipated operational and accidents). Figs. 1 and 2 show the basic approach. An axial convection (Fig 2) or conduction (Fig 1) model is used for relating T_{avg} and T_{out}. Also the "uniform pool temperature" assumption (which implies $T_{avg} = T_{out}$) is available as an option.

Then, the overall energy equation of the pool (see Fig.1,2)

$$MC_p \frac{dT_{avg}}{dt} = Q(t) - GC_p K(T_{out}(t) - T_R) \qquad 1$$

was shown in the course of our study[9] to have a general solution of the form

$$T_{avg} - T_R = (T_0 - T_R)\ \beta_{L=H}(t) + \int_0^t dt'\ \beta_{L=0}(t-t')Q(t')/MC_p \qquad 2$$

where the terms of the rigth side mean the heat-exchanger and pool mass cooling effect on the initial temperature, T_0, and additional heat load $Q(t)$. The conservative assumption of a short fuel as compared to the pool level has been made in eq 2.

The Green funtion $\beta_L(t-t')$ represents the impact on T_{avg} at time t of a heat load unit per unit mass deposited at the ground pool floor at time t', with the pool initially at the heat sink reference temperature T_R. The β_L function is specific of a given cooling system, it is different for each event type (heat exchanger or pump failures, ruptures of the type of Fig. 1 etc.) and is used to compare different cooling system designs. Typical examples are shown in Fig. 3 for single events.

Fission products decay loads (*) are calculated[7] according to

$$Q(t) = \sum_{i=1}^{N} \int_{0}^{T_i} h(t+t_{\Delta} + \tau_i + T_i - t')\, P_i(t')\, dt' \qquad 3$$

for a discharge of N fuel batches that have been irradiated with a power history $P_i(t)$, per batch. The universal function $h(t-t')$ is the decay heat produced at time t by each fission power unit generated at time t' . It is related to the $\frac{P_\infty}{P_0}$ function of ANSI-5.1[8] and A.P.S.B 9.2 technical position of SRP through[7]

$$\int_{t}^{\alpha} h(t')dt' = P_\infty(t)/P_0 \qquad 4$$

In fact, eq 3 reduces to the standart approach for constant operating histories.

In calculating Q, some care should be exercised with the $P_i(t)$ used, to take into account that central fuel elements operate at a different power than peripheral ones. As it is well known, for short cooling times only the last portion of the operating history is relevant, and discharged fuel tend to operate its last cycle in central positions.

If boiling is reached at time t_B the level changes according to

$$L(t) = L_0 + 1/\rho\Sigma \int_{t_B}^{t} dt' \left[Q(t')/h_{fg} - W(t')\right] \qquad 5$$

where h_{fg} is the amount of energy absorbed per evaporated water mass unit, $W(t)$ is the water addition rate capacity at time t of the make-up system and L_0 is the initial level, as stated in the technical specifications, necessary to assure the "minimum

(*) Heavy elements contribution is treated in the standart way (5).

shielding water depth" requirements for normal operating conditions.

The heat-exchanger type and behavior is also incorporated into the model through the expression,

$$K = \frac{\exp\{\frac{UA}{GC_p}(1+\frac{G}{G_S})\} - 1}{\exp\{\frac{UA}{GC_p}(1+\frac{G}{G_S})\}(1+\frac{G}{G_S})} \qquad \text{(Parallel Flow)} \qquad 6\text{-a}$$

$$K = \frac{\exp\{\frac{UA}{GC_p}(1-\frac{G}{G_S})\} - 1}{\exp\{\frac{UA}{GC_p}(1-\frac{G}{G_S})\} - \frac{G}{G_S}} \qquad \text{(Counter Flow)} \qquad 6\text{-b}$$

which follows directly from the classical logaritmic approach for heat exchangers of constant heat transfer UA coefficients. The usual corrections for accounting deviations from this non-realistic model are applied to the UA factors.

A code has been written to implement these equations into an automatic numerical scheme, and an evaluation procedure is being implemented to perform the numerical aspects of the SFPCS safety review in an unified way.

b) Calculation of maximum temperaturess and temperature delays.

To clarify the content of eq. 2, mayor contribution of this paper, consider a "two-ramp" decay heat load as shown in Fig. 5.a. It is simple to show that under the assumption of "axially uniform pool temperatures", i.e.

$$T_{avg}(t) = T_{out}(t)$$

eq. 1 implies that the maximum temperature (see Fig. 5.b)

i) is enveloped by the curve* (see Fig. 5.b)

$$T_{avg}(t) = T_R + \frac{Q(t)}{GC_p K} \qquad 7$$

ii) is reached with a delay* Δ , relative to the time t_o at which $Q(t)$ is maximum. This delay is due to the pool mass "temperature inertia" effect. In fact when the mass is very small, eq.1 gives $\Delta=0$, i.e.

$$\max T_{avg} = T_R + \frac{\max Q}{GC_p K} \qquad 8$$

which is the "static" approximation used by some authors that conservatively neglect the pool effect.

It has also been found[9] that i) and ii) remain valid in the forced convection case provided the approximation

$$|\ln(1-K)| \simeq \frac{K}{1 - K/2} \qquad 9$$

is reasonable, but the predictions for Δ are different and lower (conservative) than for the uniform model. In fact the effect of pool mass on the delay Δ is well approximated by

$$\Delta = \Delta_r \frac{M}{G}$$

$$\Delta_r = \frac{1}{\lambda} \ln X \qquad 10$$

with

$$\lambda = \begin{cases} K & \text{For the uniform model} \\ |\ln(1-K)| & \text{For the convection model} \end{cases} \qquad 11$$

---

* Actually these conclusions are valid for any type of heat load in the uniform model.

and γ is weakly dependent on ξ,M, and mainly dependent of the "shape" of the heat load ramps.

It is concluded from Fig. 7 that for large ξ values ("good" heat-exchanger cooling) the difference between the uniform and convection models is higher and for large γ values ("sharp" heat load shapes i.e. short discharge and cooling times) the delay difference is larger.

Delay times range from a few hours to a few days and decrease the maximum temperature by up to ten percent (in some cases) of the "static" values of eq 8 which remains a reasonable approach in many cases. Note that although these conclusions are only for "two ramp" heat loads, they are a very good approximation because it is very possible to approximate with "two ramp" shapes the actual heat loads (see Fig 6) until the maximum temperature is reached.

Figs. 8, 9, 10 show some transient results for normal conditions and two-ramp heat-loads.

From a licensing point of view the interest of eq. 10 is in determinig alarm set-points for stopping the discharge, taking into account the expected additional temperature increase.

From a design point of view credit for the reduction in max T_{avg} due to the delays may be used to alleviate the cooling system requirements, because design heat loads are rarely approached and normal margins are very high (peak design type).

c) Examples of transient temperature results.

The figures following Fig. 10 show some of the results obtained with the evaluation code PISCAL. They represent normal and off-normal conditions under the heat loads presented in Fig. 6.

As may be seen, for normal conditions the difference between T_{avg} and T_{out} can be substantial. For off-normal conditions, the key parameter is the time before boiling. It turns out that a condition II event, like a "one of two pumps failure" may be more limiting than a loss of all pool cooling if sufficient - make-up capacity is available.

The "rupture of inlet pipe" event, is still being under study to include effects of natural convection. It should be noted, however, in the context of this paper, that the natural convection in the pool is of "enclosed type" around the fuel, and the water mass "affected" by the fenomenon changes with time. This justifies not to include it (conservative) in normal conditions, from the - point of view of SFPCS. But for an abnormal event like this rupture, we need more study.

In some cases, we will need to consider the effects of loss of pool water. We are implementing for this purpose the techniques of Ref. 12.

V. Comments and Conclusions

a) Comments.

The general problems of the safety evaluation of nuclear installations in Spain have been described in Ref. 10. The sections before have detailed some of them for our particular system, like - the difficulties of "importing regulations" and the necessity of developing clearly stated regulatory positions. Other aspects of interest will be commented here.

The concept of a "reference plant" evaluation method, as commonly understood, is only useful, when applied to the SFPCS, in minor secondary points like comparing safety classification component tables, type of equipment supplied and similar "hardware" aspects. For the software, where "numbers" are of interest, the concept is only meaningful if it is understood in a sophisticated way, namely, the comparison of a certain "key parameter list" with stated conservative directions. But, how to prepare such a list, without a deep generic study on the subject? and if such a study is not generated indoor, how to apply the list without a "dangerous" and maybe impossible interpretation of the conclusions of "foreign generic studies" never sufficiently detailed and mostly unavailable?

In fact, in developing the evaluation model of section IV, we found interesting parameter combinations to make a reasonable comparison, like the "beta functions" of the cooling systems, but we doubt of anybody that could judge on the basis of a "key list" without knowing details of the evaluation model. For instance, - different heat loads could clearly counteract any advantage in "beta functions".

Actually, what is understood by "independent verification" (as the application of the evaluation model), turns out to be the same as this sophisticated interpretation of the "reference installation". It is our experience that more than the comparisons with "other's calculations" is the intercomparison between "different installations calculations", what gives a basis for judgements that, at the end, should be flexible and more cualitative than rigidly cuantitative.

In summary, by developing the model it is possible to "rationalize" the engineering judgement that is essential to licensing action. This also puts in perspective the relative value of numerical analysis which shoud never be overemphasized.

From the point of view of planning and organisation, it is important to quote that the development of a generic study in this relatively minor problem, took about six months-man effort, that were made under assignement to a particular installation because our organization do not recognize yet the need for generic studies. However with the aid of the study we estimate now two weeks, at most, in performing an application to a particular plant.

Finally, we consider essential to contact owners and suppliers and discuss with them the implication of any licensing action. In many cases, particular details of an installation make "unreasonable" a decision that would be otherwise a logic conclusion of the safety philosophy. From this point of view and contrary to what happens in NSSS related tasks, we have not found any problem and the channel communications were good enough.

b) Conclusions.

An evaluation model has been presented for establishing regulatory requirements in the SFPCS system. The mayor design factors, regulatory and design limits and key parameters have been discussed. A regulatory position for internal use has been proposed. An associated numerical model has been developed, the salient features of which have been shown. A simple formula for calculating temperature delays and maximum temperatures under normal conditions was presented. Finally, associated problems and experience were discussed.

AKNOWLEDGMENTS

It is a pleasure to recognize the excellent effort made by Mr. Conrado Prieto in performing the FORTRAN Programming and handling of the J.E.N "UNIVAC 1106" computer machine. Also the careful type-writting by Miss Angeles Feria and the illustrations by Mr. Félix Ancos are greatly appreciated.

COMPLAINS

We apologize not to have the time necessary to make this paper shorter, as well as the necessary foreign languages knowledge.

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TABLE I (1 of 3)

SUMMARY OF U.S.A. REGULATORY AND INDUSTRY REQUIREMENTS RELATED TO SFPCS

‡ INDICATES SOME AMOUNT OF INTERPRETATION ADDED TO THE WORDING

SOURCE AND DATE	APPLICABILITY	EVENT CLASSIFICATION	T_{avg}^{*} LIMITS	LEVEL LIMITS	RELIABILITY REQUIREMENTS OF PROTECTIVE MEASURES {and/or} SOMMENTS
Standart Review Plant (SRP) from rev o (Fef 1975) up to rev 3 (1979) and 10CFR50 ap.A General Design Criterions (GDC)	Light water plants after 1975. (depending on revision)	**Normal Conditions** (NC) "...Expected necessary manoeuverings"	T_{avg}^{*}<140ºF one cooling train‡ normal heat load‡ T_{avg}^{*}<140ºF two cooling trains‡ maximum heat loads‡ (heat loads calculated with BTP APSB 9.2)	Maintain 10CFR-20 Exposure and radiation limits (0.12% failed fuel)‡ and "specified acceptable fuel design limits"	FSAR's used: T_{avg}^{*} < 150ºF Maximum heat load two cooling trains; T_{avg}^{*} < 125ºF Normal heat load one cooling train. Accepted by NRC reviewers (prior to 1976)
		Anticipated occurrences (AOO) "...Incidents occurring once in the plant life..."	Boiling‡ not allowed	MAINTAIN 10CFR-20 Exposure and radiation limits (1% failed fuel)‡ and "specified acceptable fuel design limits"	"Suitable redundancy of components so that safety functions can be performed assuming single active failure of a component, coincident with the loss of all off-site power"
		Accidents (AC) "...Postulated..."	None ¿Boiling allowed?	Reg.-guide 1.13 Fuel remain covered Maintain 10CFR100 off-site exposure and radiation limits	Endorses Reg. guide 1.29 that requires "necessary portions for cooling decay heat to be seismic" (enforced by reviewers until 1977) SFPCS portions seismic

TABLE I (2 of 3)

SUMMARY OF U.S.A. REGULATORY AND INDUSTRY REQUIREMENTS RELATED TO SFPCS

SOURCE AND DATE	APPLICABILITY	EVENT CLASSIFICATION	T_{avg}^{*} LIMITS	LEVEL LIMITS	RELIABILITY REQUIREMENTS OF PROTECTIVE MEASURES {and/or} COMMENTS
ANSI-N210 (1976) REVISION 0	LIGHT WATER PLANTS AFTER 1976	Condition I "... Are operations expected frecuently or regularly ..."	Personnel confort Limits of essential equipment Limits of instruments $T_{avg}^{*} \leq 150$ºF at full storage	Minimum shielding depth, Allow fuel handling. Maintain 10CFR20 with* 0.1% fuel leakage (2.5 mrem/hr)	They are initial conditions for other events
		Condition II "... Include incidents ... which may occur during a calendar year"	Boiling * not allowed	Minimum shielding depth. Allow fuel handling, Maintain 10CFR20 with* 0.25% fuel leakage	Single active failure Different terminology and definitions than in GDC
		Condition III "... Include incidents ... which may occur during the lifetime"	Boiling allowed?	Minimum shielding depth (unspecified ??). Allow fuel handling. Maintain 10CFR100 (100%!!)	Single active failure GDC requires 10CFR20. SFPCS Safety class 3 (article nº 4.3.1) but exceptuated !! in article nº 5.1.9.3.2
		Condition IV "... Are faults nor expected ... but postulated..."	Boiling allowed as an alternative	Minimum depth for fuel heat transfer. Maintain 10CFR100	Single failure. Safety class 3 Redundancy, seismic category I (if protectiong SSE, OBE) An alternative allows SFCS not to be seismic

TABLE I (3 of 3)

SUMMARY OF U.S.A. REGULATORY AND INDUSTRY REQUIREMENTS RELATED TO SFPCS

SOURCE AND DATE	APPLICABILITY	EVENT CLASSIFICATION	LIMITS	LEVEL LIMITS	RELIABILITY REQUIREMENTS OF PROTECTIVE MEASURES {and / or} COMMENTS
International[11] storage meeting (NRC sponsored) (1977) W. Butler lecture (NRC)	Modifying Existing Facilities	Endorses ANSI-210	Endorses ANSI-210 but $T_{out} < 150°F$ not $T_{avg} < 150°F$	Endorses ANSI-210 with the "Boiling allowed" alternative	It is considered to represent actual practice within NRC review process (heat loads calculated with APSB-9.2 technical position of SRP)
Same meeting "Proposed position for review and acceptance..." D. Eisenhut lecture (NRC)	Unknown	Do not talk about A O O events	Endorses ANSI-210 except $T_{avg} < 140°F$ (SRP)	Boiling not allowed* Alternate cooling required under accidents	Status Unknown Suggest tec-spec on temperature limits. Seismicity of alternate cooling implicitly required (makeup not mentioned)
Enviromental[6] impact statement (NRC) August 1979	Design basis not regulatory requirements	Same as SRP	140-150°F with normal cooling 125°F not a requirement		SFPCS components Classified as "seismic category 2"?? ANSI-210 and SRP endorsements referred simultaneously

TABLE II

PROPOSED INTERNAL POSITION (J.E.N.)

APPLICABILITY	EVENT CLASSIFICATION	T_{avg} LIMITS	T_{out} LIMITS	LEVEL LIMITS	RELIABILITY REQUIREMENTS OF PROTECTIVE MEASURES
LIGHT WATER PLANTS	Normal conditions "... operations expected frecuently or regularly..."	≤150ºF One cooling train. Normal heat load. ≤150ºF Two cooling trains. Maximum heat load. Limits of instruments. Limits of essential equepment.	140ºF One cooling train. Normal heat load. 140ºF Two cooling trains. Maximum heat load. 125ºF Two cooling trains and normal heat load (not strictly required but "nice to have"	Maintain minimum shielding depth (with 0,12% of fuel leakage) within 10CFR-20 exposure limits (2,5 mrem/hour) allowing for fuel handling movements	
	Abnormal conditions "...incidents expected to occur once in the life of the plant..."	Boiling not allowed	Tout <212ºF	Maintain minimun shielding depth with additional 1% fuel failures and initial 0,12% fuel failures within 10CFR-20 exposure limits (2,5 mrem/hour) allowing for initial fuel handling movements	Corrective measure able to perform its safety function including single active failure. Safety class 3 required.
	Accidents "... faults not expected to occur but postulated..."	Boiling allowed		Maintain the fuel covered, and radiation and exposure within 10 CFR-100 limits with initial minimum water shielding depth as in normal conditions.	Corrective measure able to perform its safety function including single failure. Seismic category 1 required if protecting SSE, OBE, consequences. Safety class 3 required

$$\frac{\delta}{\delta t}\int_V \rho\, h\, dV + \int_S \rho h\vec{v}\cdot d\vec{S} - \frac{\delta}{\delta t}\int_V P\, dV - \int_V (\vec{v}\cdot\vec{\nabla} p)\, dV = \int_S \vec{q}\cdot d\vec{S} - \int K\, \vec{\nabla}\vec{T}\cdot d\vec{S}$$

$$M C_p \frac{\delta \overline{T}}{\delta t} + G\,(h_{out} - h_{in}) - \quad 0 \quad - \quad 0 \quad = Q(t) - \text{HEAT CONDUCTION}$$

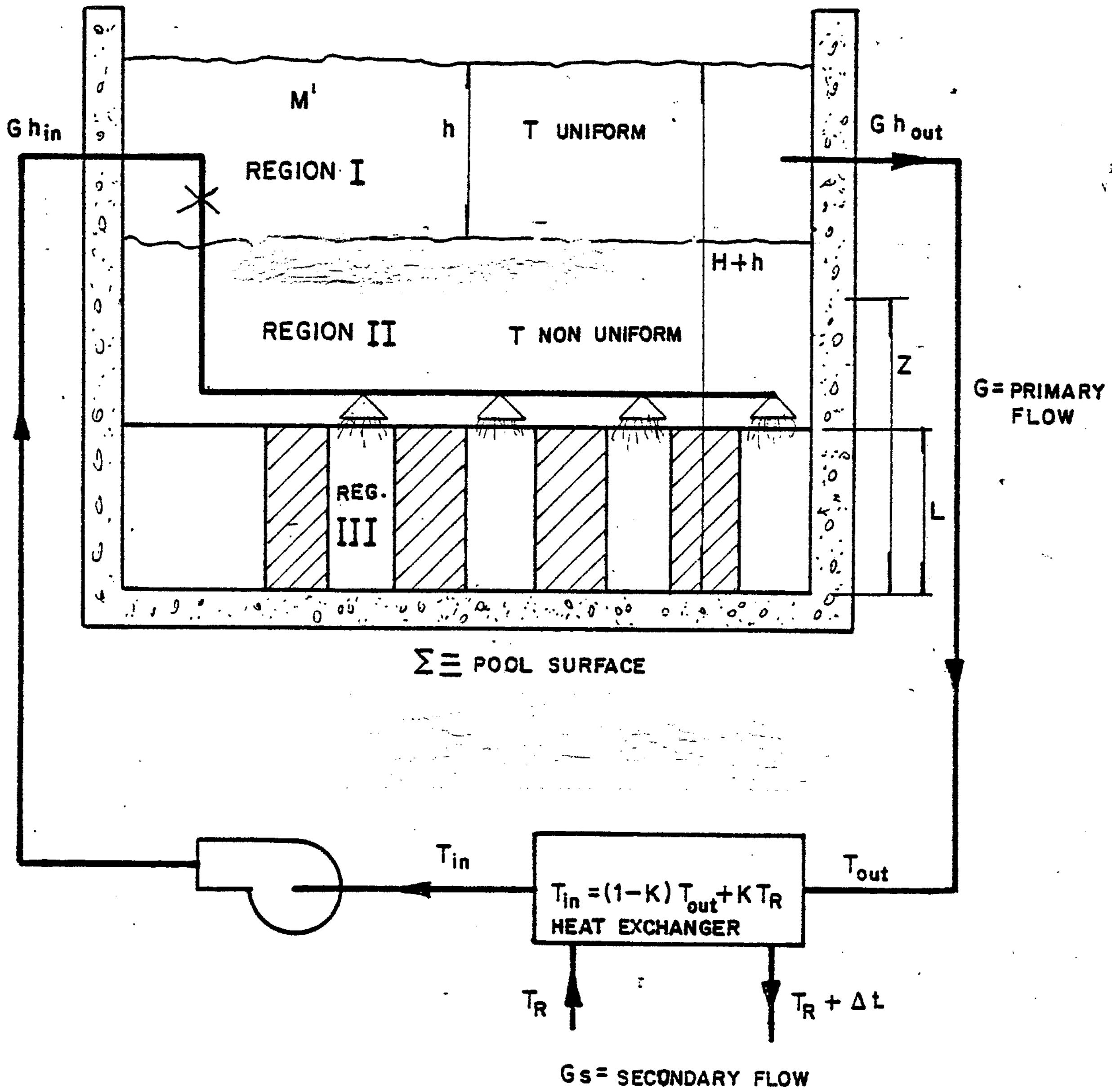

FIG.1.— RUPTURE OF AN INTERNAL INLET PIPE.

$$\frac{\delta}{\delta t}\int_V \rho\, h\, dV + \int_S \rho h\vec{v}\cdot d\vec{S} - \frac{\delta}{\delta t}\int_V P\, dV - \int_V (\vec{v}\cdot\vec{\nabla} p)\, dV = \int_S \vec{q}\cdot d\vec{S} - \int K\,\vec{\nabla} T\cdot d\vec{S}$$

$$M\, C_p \frac{\delta \bar{T}}{\delta t} + G\,(h_{out} - h_{in}) \quad - \quad 0 \quad - \quad 0 \quad = \quad Q\,(t) \quad - \quad 0$$

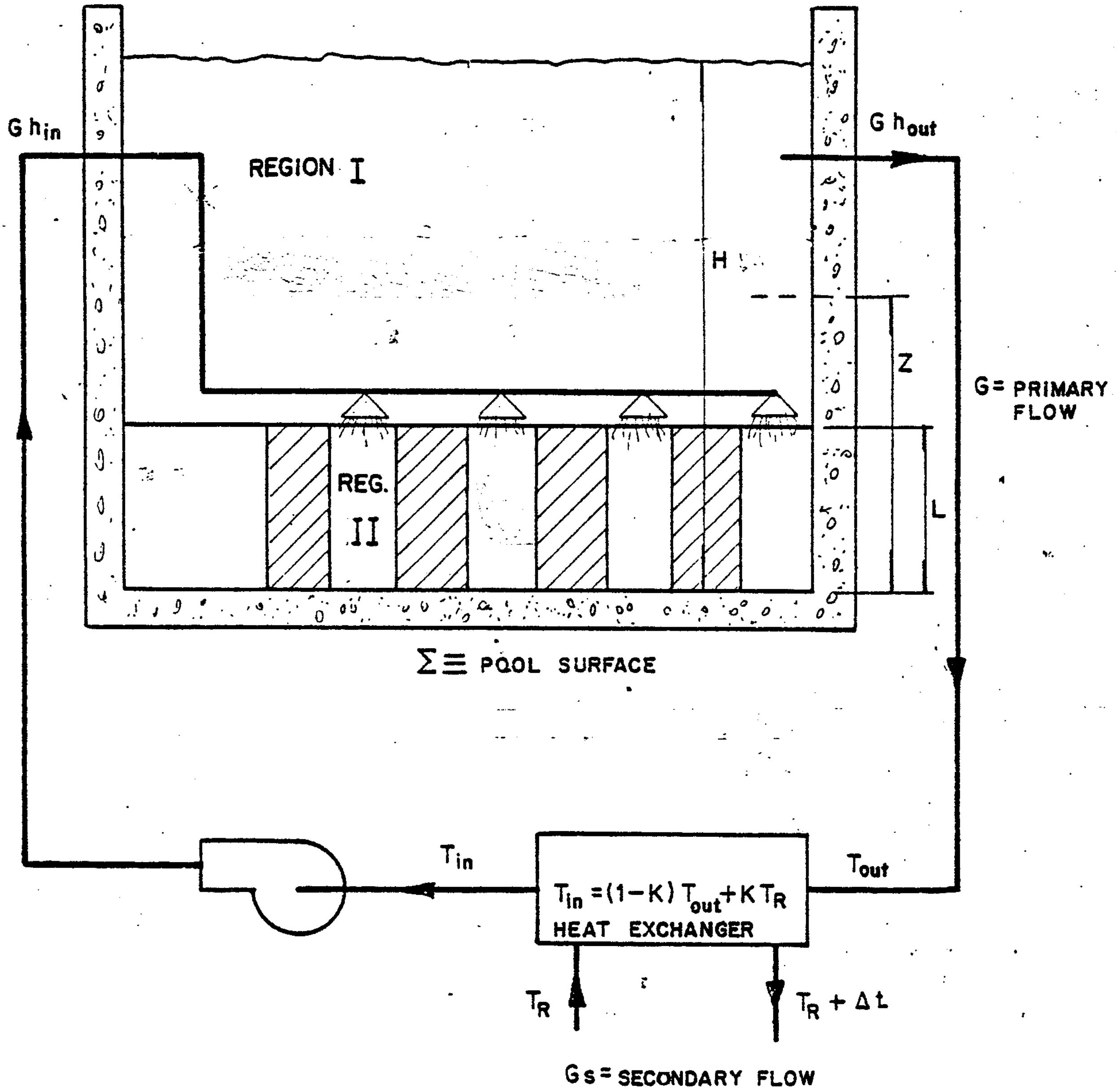

FIG. 2.— FORCED CONVECTION POOL COOLING MODEL.

DSN/3003a/1079

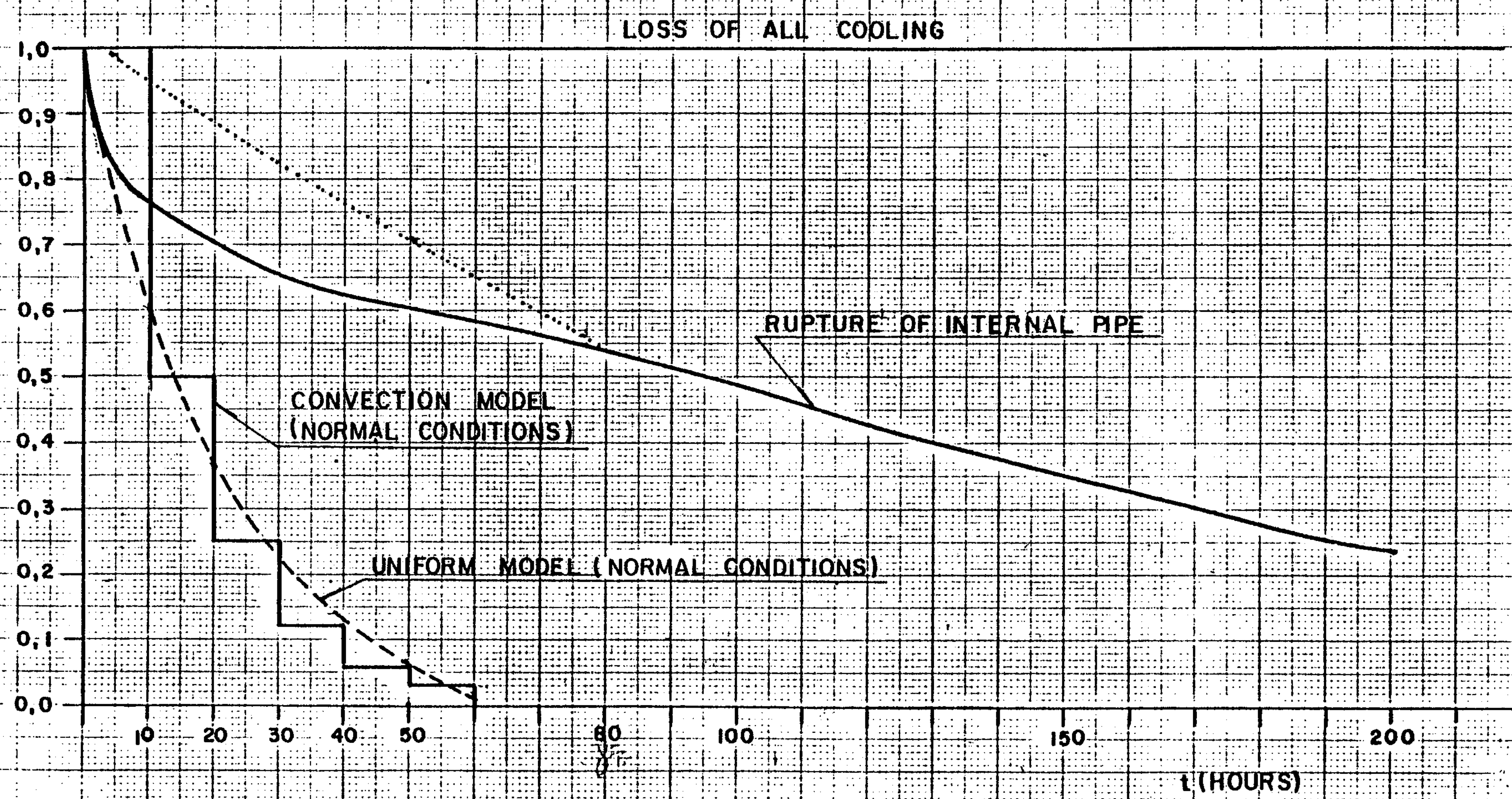

FIG.3.- SINGLE EVENTS BETA FUNCTION.

DSN/3012a/1079

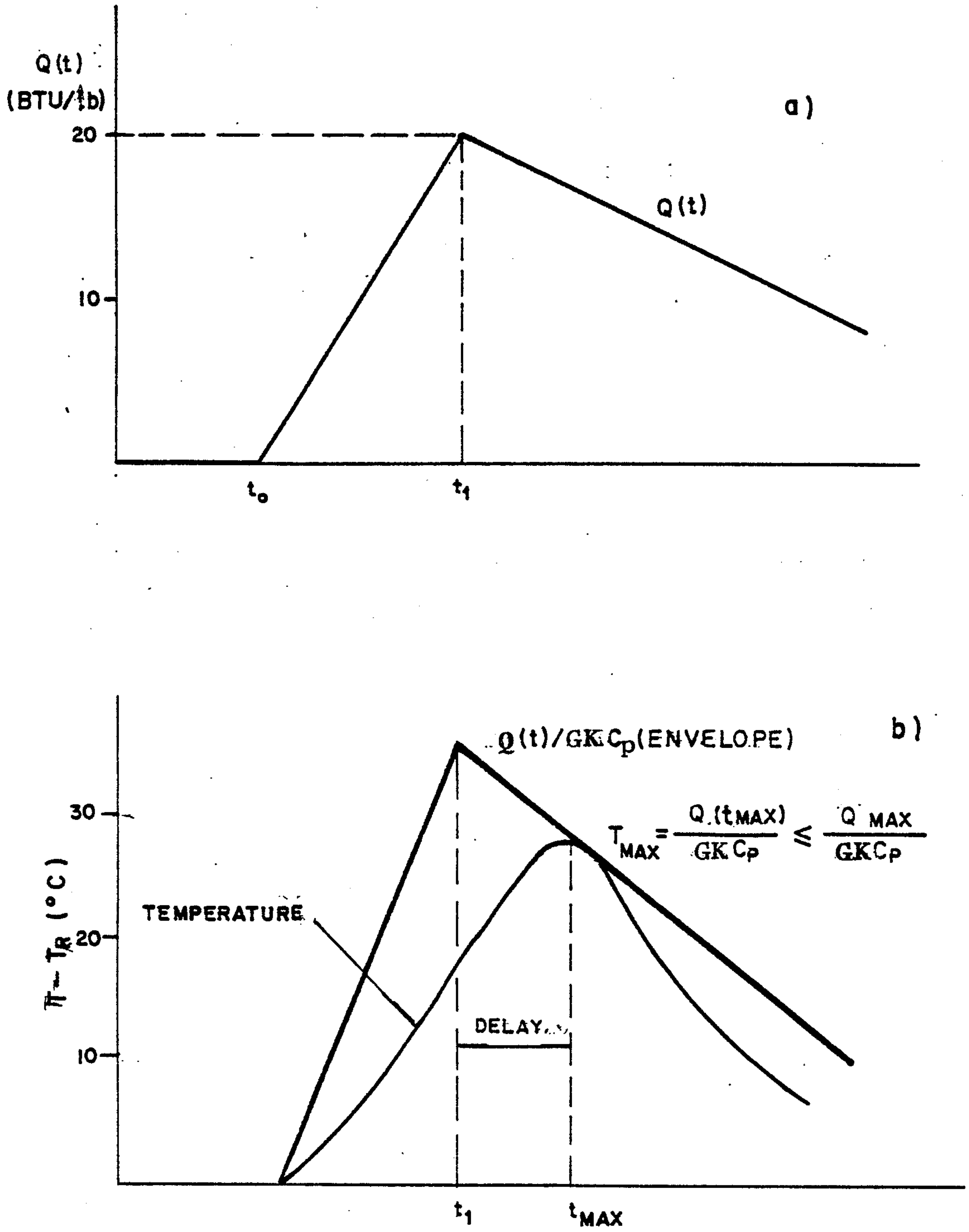

FIG. 5.— MAXIMUM POOL TEMPERATURES FOR TWO-RAMP HEAT LOADS.

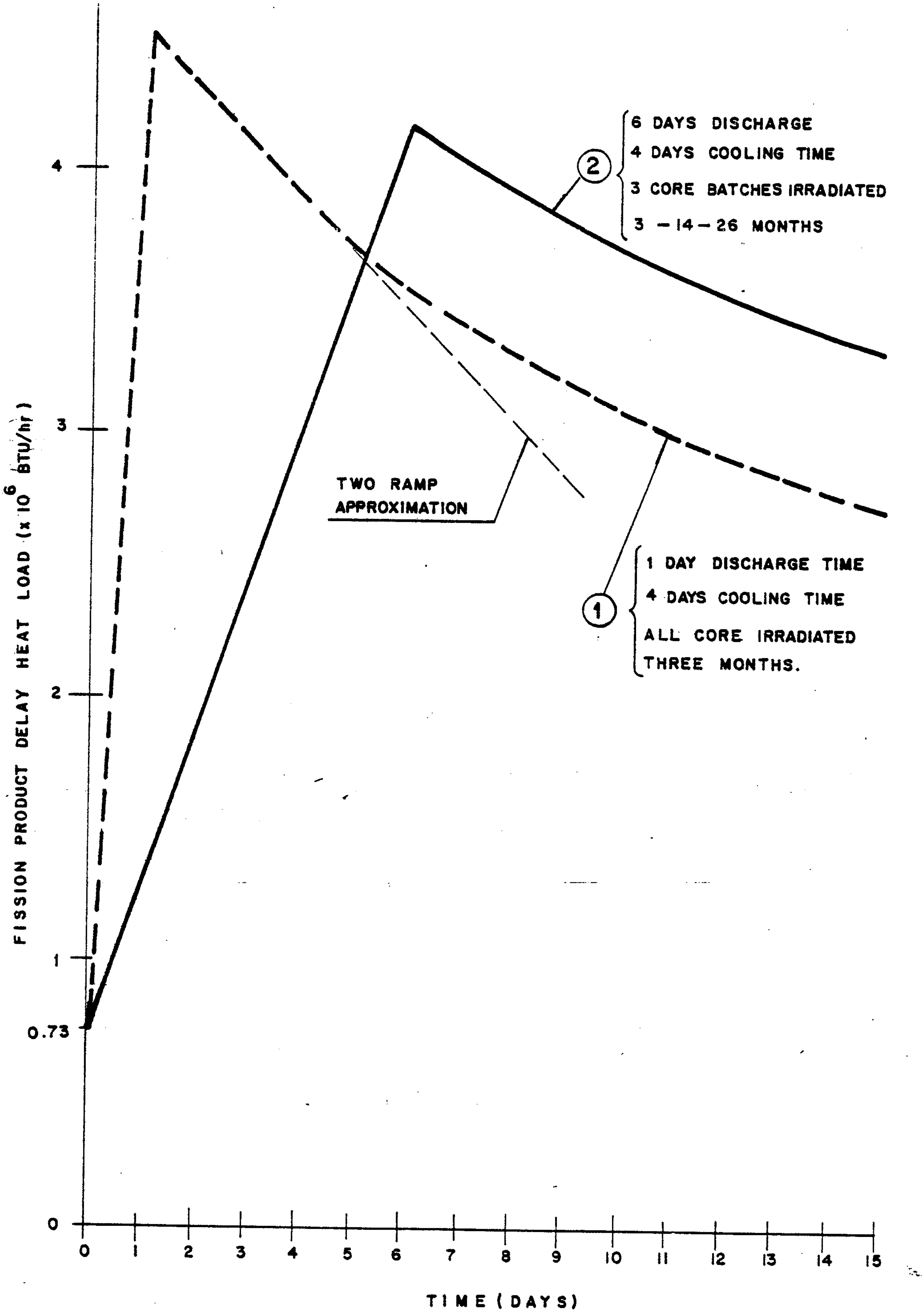

FIG. 6.—FISSION PRODUCT DECAY HEAT LOADS.

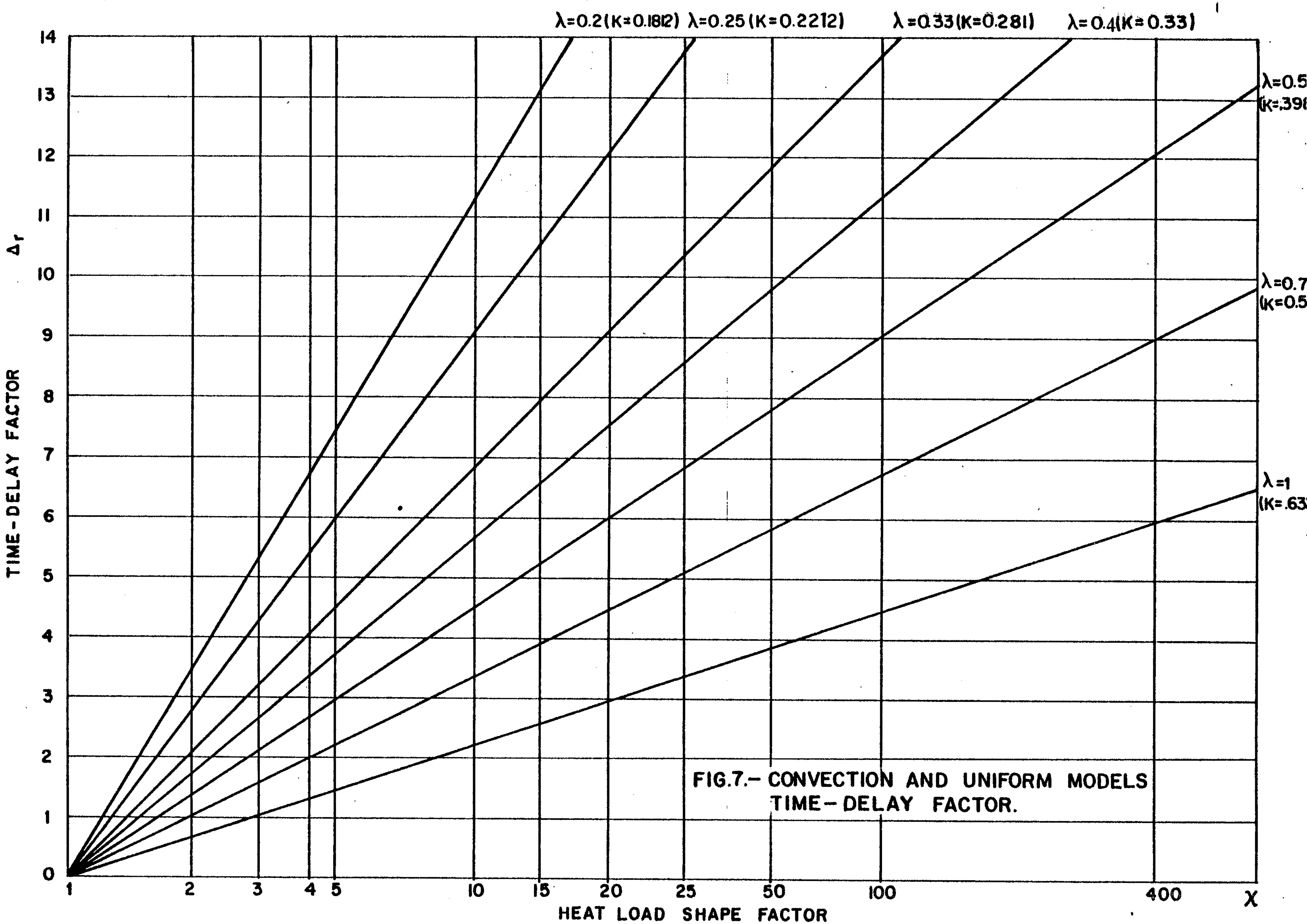

FIG.7.– CONVECTION AND UNIFORM MODELS TIME–DELAY FACTOR.

DSN/3026/1179

DATA:

$G = 5 \times 10^4$ Kg/hr $M = 5 \times 10^5$ Kg/hr $Q_0 = 5 \times 10^5$ Kcal/hr

$Q_1 = 5 \times 10^2$ Kcal/hr $K = 1/3$ $T_0 - T_R = 7°$ C $T_0 = 23°$ C

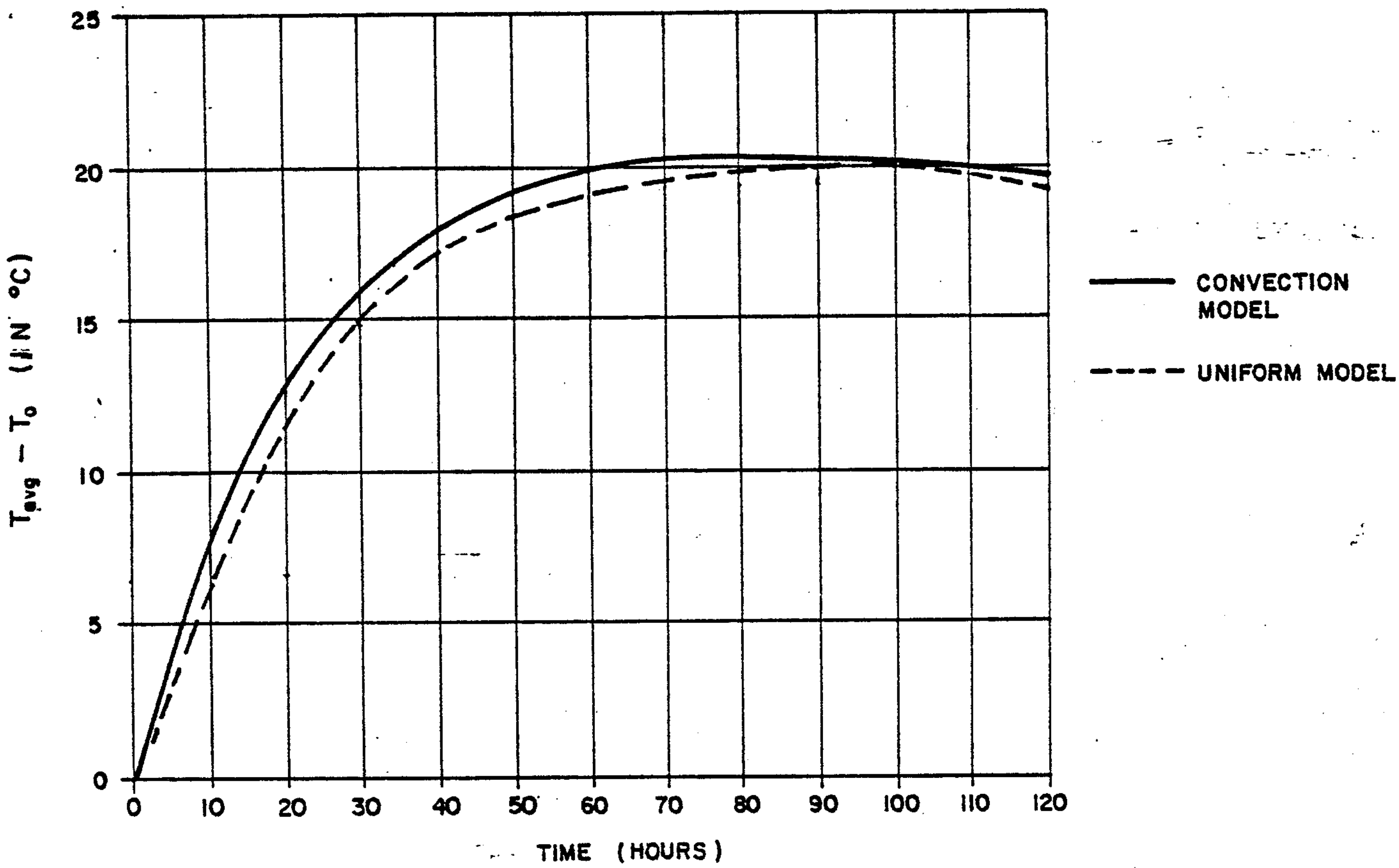

FIG. 9.—NORMAL DISCHARGE (RESPONSE TO A RAMP $Q_0 - Q_1 t$)

DSN/3014a/1079

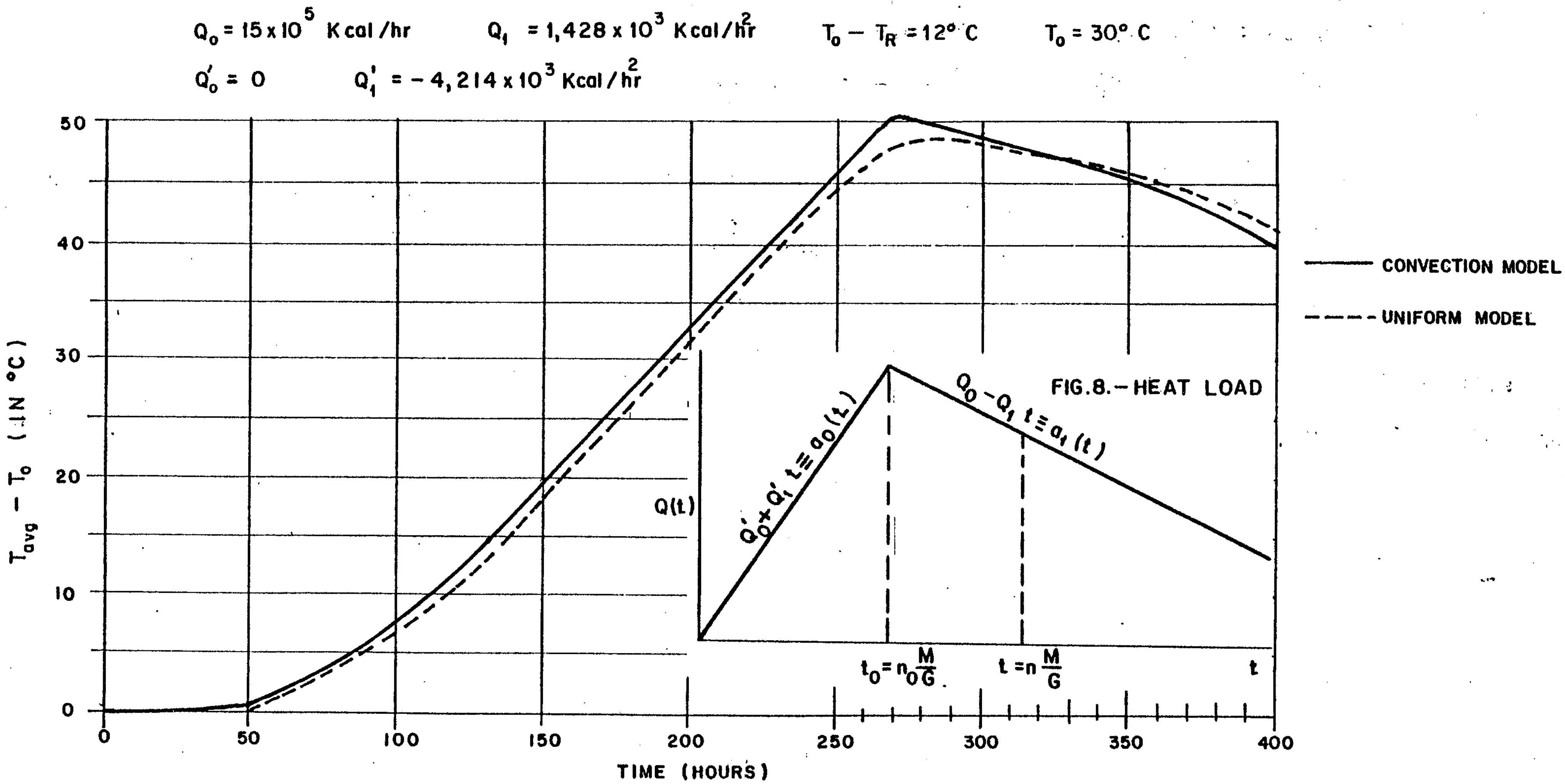

FIG. 10.– TOTAL DISCHARGE OF CORE IN 11 DAYS.

DSN/3015a/1079

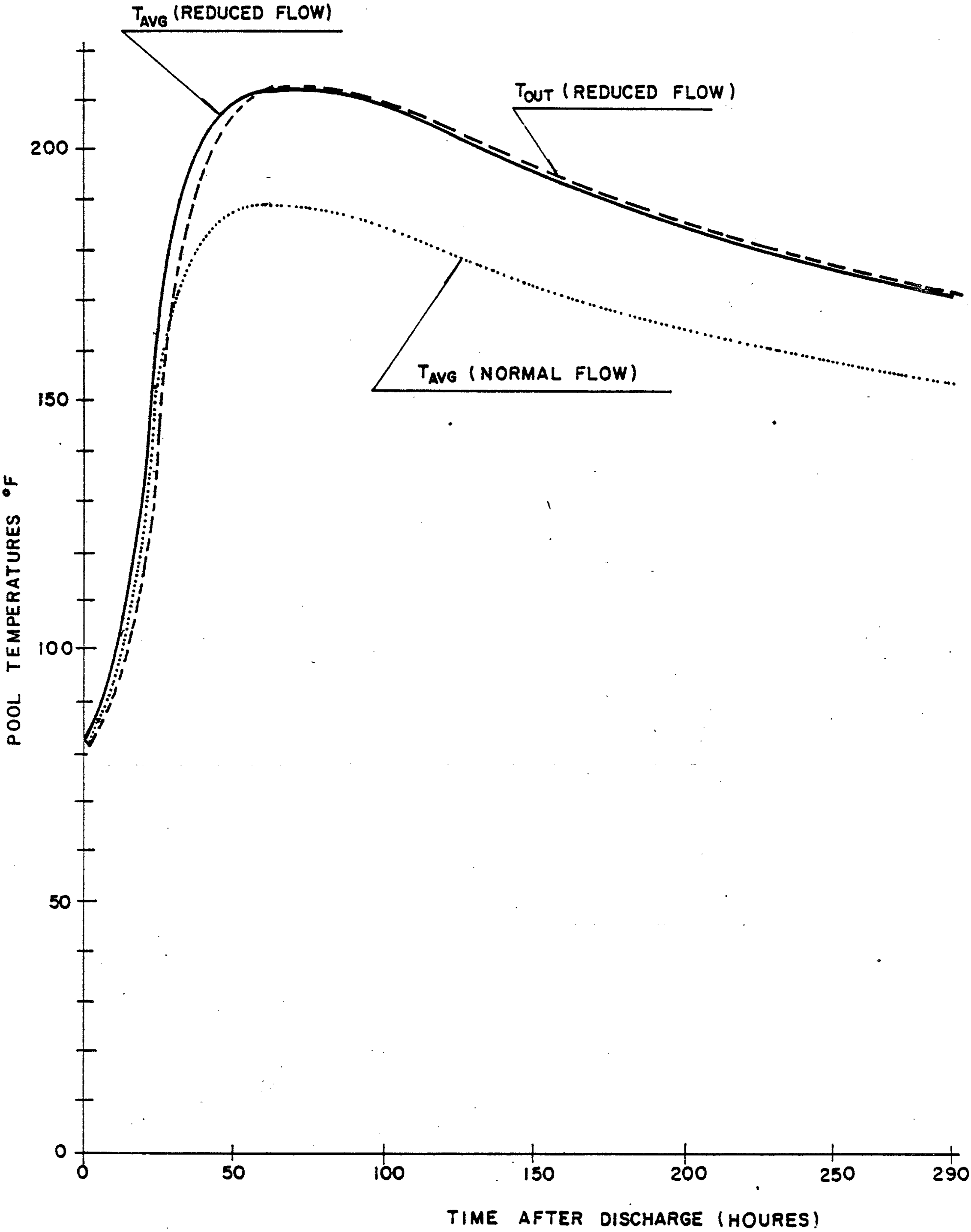

FIG.11.— NORMAL CONDITIONS.

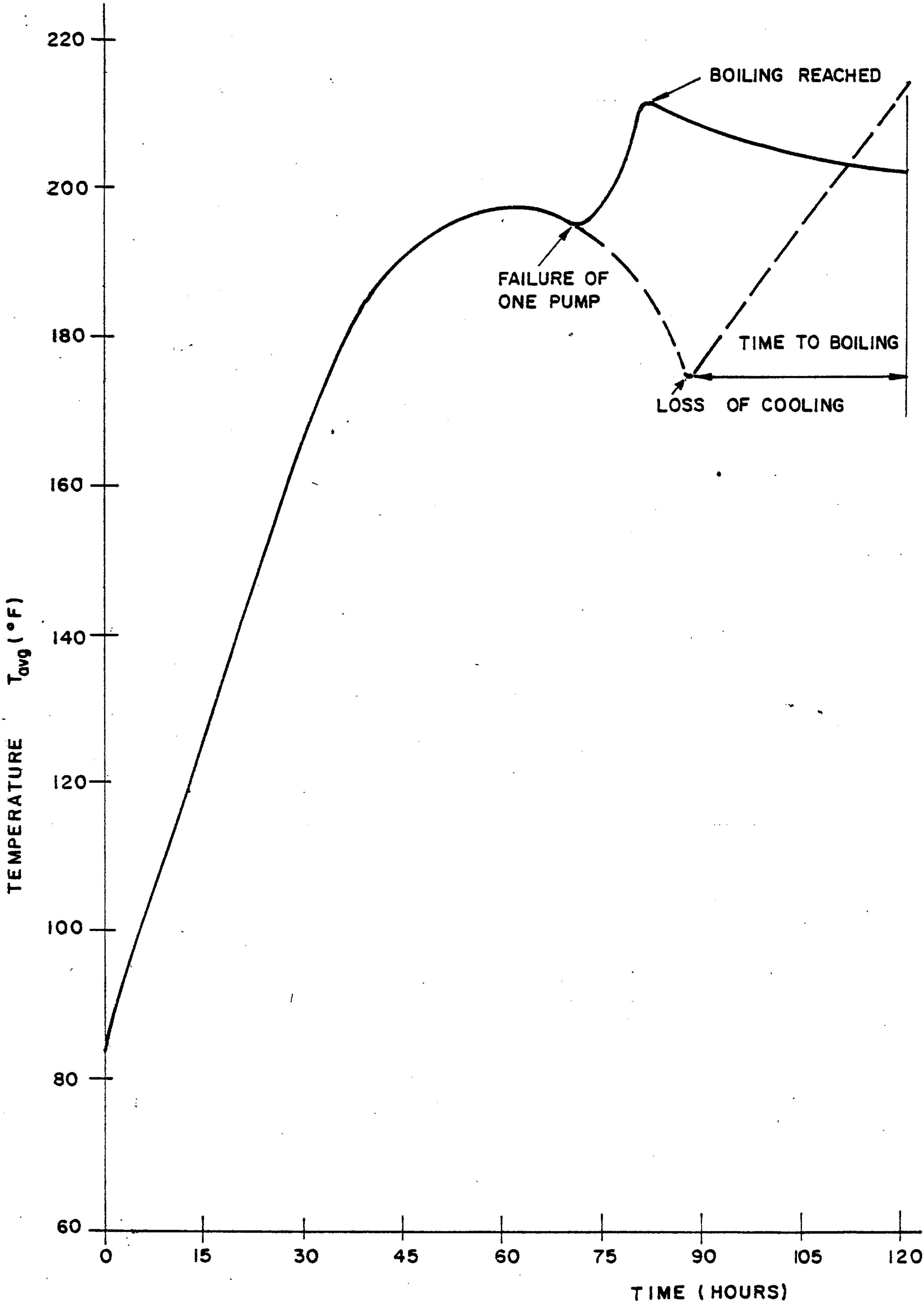

FIG. 12.— OFF-NORMAL EVENTS.

Third Period of Discussions on Session VIII

P. Giuliani (Italy) (To R. Solá)
Are you developing in Spain safety guides in the site analysis area?

R. Solá (Spain)
Up to now no guidelines on this subject have been developed in our country, but we hope that in the near future this type of activity will be undertaken.

N. Aybers (Turkey) (To J. Izquierdo)
NRC Reg. Guide 1.29 requires that the spent fuel pool must be seismic category I, also the cooling circuit must be the same and an extra make-up line is required. What do you think about these requirements.

J. Izquierdo (Spain)
Our position implies an analysis for classifying earthquakes in the different categories, and the seismic requirements are then a consequence of it. The entire make-up system should of course be seismic. If necessary we accept alternate seismic cooling systems provided that analysis shows its adequacy and availability for the postulated situations. Only if this is not possible, the SFPCS is required to be seismic. For the pool itself, it should be seismic to the extent necessary to ensure fuel cooling and fuel geometry. For old plants, if fracture of walls can be given credit, a detailed analysis of the loss of pool water accident should be performed and adequate protection provided to maintain the radiological protection criteria for accidents and the fuel integrity. The extra-make-up line is required under maintenance of the single (active or passive) failure criterion of the protective measures under accidents.

J. Laaksonen (Finland) (To J. Izquierdo)
In your proposed internal position you allow boiling of water in accident conditions. Have you analyzed the effect of hot water to pool structures and do you think that your concrete can for a long time stand that high temperature without losing its integrity?

J. Izquierdo (Spain)
We did not do any 'ad-hoc' study. The information supplied by our structural experts indicates that for quarz-concrete pools under sealed or dry conditions, there is no loss of

compression strength at extended high temperature time intervals. Although for limestone concrete under sealed conditions some loss is expected, the situations for which a substantial pool water leakage could occur are far away. Liner steel oxidation is only expected at highly localized junction points. On the other hand, notice that 'boiling or not boiling' does not make much difference about temperatures, but rather is a matter of water evaporation rates and pool liquid-water levels. Finally, our position will be subject to comments of the structural experts before approved.

SESSION IX

CONCLUSIONS AND RECOMMENDATIONS OF THE SPECIALIST MEETING

Chairman: P. Giuliani

Scientific Secretary: J.P. Olivier

Panel Members

P.B. Woods

G. Tenaglia

J. van Daatselaar

K.B. Stadie

J.P. Olivier

P. Giuliani

Panel Members

P. Giuliani

R. Gausden

P. Govaerts

S. Israel

A. Jensen

Introductory Remarks by the Chairman

P. Giuliani

This is the last session of this meeting, Session IX. We've had a number of rather interesting sessions, and the chairmen or secretaries of the different sessions are now going to make a very short resumé of what was said during their sessions, and we'll point out the highlights; then Mr. Stadie will say a few words on behalf of the NEA, and we'll open the discussion.

I was proposing to do myself a very short resumé, but since all the chairmen are going to have theirs, let's save some time by starting with the chairman of the 2nd Session.

SESSION II

Mr. P. Woods

The Chairman of the Session, Dr. Pfaffelhuber, asked me to give you his apologies as he is not able to be with us today.

Just to remind you, the Session dealt with organisational matters and there were five papers presented verbally and a written paper received from Turkey.

This session was opened by Mr. Gausden, who described features of the organisation and experience with the regulatory review in the UK. His account of the stages of the review brought out the similarity of approach between countries, even though they may have arrived at the requirements independently. He emphasised the need for a continuing review as plants aged and approached their design life.

Mr. Gausden also touched on the question of making review information available to the public and this was taken up by Mrs. Perez in her very comprehensive presentation of organisation and practices in review and licensing of nuclear power plants in Spain. On this point she concluded that free access to files and data should be provided where possible.

Mrs. Perez also discussed the problems brought about by scarcity of trained and experienced manpower which she considered to be one of the stumbling blocks to the safe implementation of the Spanish nuclear power programme. This same point was made by a number of speakers and is also referred to in the paper "Regulatory Organisation in Turkey", which was received but not read. There were problems in obtaining suitable numbers of staff, providing training and experience and support in terms of outside consultancies and research activities.

The paper by Mr. Laaksonen on the structure, qualifications and training of the regulatory body staff in Finland was encouraging in that it showed how the problems which have to be faced when first developing nuclear power might be overcome, even in the case of a country as small as Finland. He dealt with qualifications and training and the special fields of responsibility felt necessary for the reactor safety department, and he emphasised the importance of combining experience of duties as reviewer and as an inspector as an effective means of acquiring the necessary in-depth understanding.

In France, a rather special view is taken on standardisation,and Mr. Oury presented an interesting account of the regulatory system in that country with its emphasis on authorisation related to standardisation. Whilst it was accepted that there are detailed matters which require the regulatory inspectors to fall back on their judgement and training, the standardisation of nuclear power plant design which is possible in France (but not always in other countries) clearly has safety advantages, for example in permitting in-depth study of specific safety problems and in achieving more efficient use of available regulatory effort. Other countries may well look with envy at the French situation.

Dr. Matulla completed the session with his review of the regulations and licensing process developed and used in Austria from 1971 until the referendum of 1978 following which the industrial use of nuclear energy was prohibited. He pointed to the dangers of specialists having too narrow a view, since safety problems must be seen in all their aspects and he stressed the importance of the licensing body keeping an eye on the organisation of constructors and operators to ensure that "border fields" between the two were not overlooked.

Many topics of interest were raised, some of which were discussed in detail in later sessions, due to the time available. I will confine myself to a few points which appealed to Dr. Pfaffelhuber and myself as being of widest interest.

(1) First, there was the discussion of staffing of the Regulatory Body in those countries which were just starting a nuclear power programme. How could the required expert staff be obtained, how could they be trained and what was the best way of giving them experience. For example, the importance of continuing experience in review and inspection was mentioned, but how could this be achieved before the nuclear programme was developed?

A related question was the size of the Regulatory Body and the effort required in different areas of its activities and also the support it needed from consultants and researchers. There seemed to be no clear consensus as to the optimum ratio of numbers of staff to numbers of plant under review.

(2) Second, as a matter of safety concern, should a country standardise on the reactor system to be licensed, as in France and Belgium, at least until considerable experience had been gained? This has advantages for example in permitting in-depth understanding of the plant, but perhaps there is also some virtue in diversity in case safety problems specific to the system chosen are found.

(3) Of interest in a number of countries is the question as to whether or not information on Regulatory Body activities should be made available to the public. By this I mean the detailed working and views of the Regulatory Body rather than the reporting of incidents where there is no disagreement. Should files and data relating to the review be openly available as was suggested, or would this lead to two levels of working and extra work for the regulatory staff?

(4) Finally, there was clearly an interest in the use to be made of reliability and probability analyses in the review. Can we now discard the deterministic approach in favour of probability and risk analyses? This question was discussed in greater detail in Session VI

SESSION III

Mr. G. Tenaglia

To briefly summarise the content of the discussions held during Session III, on the Technical Bases for Regulatory Review, one can refer to some important aspects of the whole problem which emerged during the discussions and require further clarification.

The first aspect concerns the value and importance to be attached to the technical regulations. If technical regulations are necessary to obtain the desired level of safety, it is not of course sufficient to guarantee that such a level will be reached.

The second aspect focusses on the necessity of having an entire set of documents, in hierarchical order, containing legislative texts, regulations, norms, etc., and in which the different levels and different technical aspects are clearly defined. In certain countries an effort is being made in this direction.

The third aspects concerns the harmonisation of the norms and regulations. An auspice in this connection was expressed by more than one speaker, but it is evident that harmonisation must pass through a clear hierarchy of norms and regulations since harmonisation is possible on some levels but not at all levels.

A fourth aspect concerns the transfer of regulations, which happens when a country which has not the same level of nuclear technology imports such technology. The acceptance of some regulations also implies the acceptance of the level of safety of the exporting country, with the possible difficulty of not being sure that the local conditions allow their full implementation.

In conclusion, the benefit of regulations has not to be proved, but one must be fully aware that their application does not allow the building of nuclear installations if it is not supported by a complete and detailed technical assessment.

SESSION IV

Mr. R. Gausden

The first paper by Mr. Maniori developed the two lines of action in the regulatory review procedures in Italy. Broadly this entailed a detailed review of the project at different stages of construction and the QA approach which concentrated on the reliability of the organisation involved and their capability to perform their duties correctly.

Mr. Gea in the second paper outlined the very considerable problems of a country such as Spain in attempting to regulate the safety of nuclear power plants without adequate resources. He outlined a "mini-analysis" type of approach. The requirements for a "proven type plant" where work planning is essential, and the establishment of basic evaluation criteria and the use of reference plants.

The final paper in this group by Mr. Jastrzeb outlined the licensing procedures in France with some interesting examples of the stages of authorisation in the construction and commissioning of Fessenheim. The clear attention to detail and the control at every stage was evident.

The UK paper concentrated on the detailed application of the licensing procedure in the UK and gave some interesting examples of problems, particularly in the construction area. Safety assessment principles have been a considerable aid to the orderly approach to an assessment spreading over several years.

Finally, Mr. Justin was concerned with Super Phénix. To some extent his approach was based on conventional methods, but with the added complication of changes to the design, some of quite a major nature as the concept evolved and construction was proceeding. A step-by-step approach leading to fuel loading in 1982 and operation in 1983 was described.

SESSION V

Mr. J.P. Pélé

During this session we had a very interesting and useful exchange of views supported by the very well documented exposé of Mr. Israel and the contributions of different countries.

Mr. Israel gave us a survey of the enormous work done by NRC and discussed the main recommendations made by this organism after the TMI accident.

The other contributions reflecting the many concerns of other countries vary with the magnitude of the national nuclear programmes and the existence of national nuclear industries.

In all the countries the TMI accident had a large impact and the national competent authorities (regulatory bodies) made a great effort to

- analyze the accident;
- take the necessary short and medium term actions;
- engage long term actions including (where applicable) new or modified research programmes.

The main items developed by Mr. Israel and picked up in the discussion are:

(1) - Emergency procedures

Lack of coordination between designers and builders on one hand and those responsible for development of emergency procedures and training of operators on the other.

In this context it was recognized that there was a need to

- study in detail the small LOCA;
- define new clear criteria for the operator during such circumstances;
- make the study of accident sequences on longer time duration (hours or days instead of minutes).

(2) Control room operation: this problem was the subject of many discussions. The main question is the communication between man and machine, but the following are also cogent:

- Presentation of process variables in normal and accident conditions;

- Degree of automatization;

- Actions to be taken by the reactor operator;

- Recognition that a compromise must be found but the solution is not yet clear;

- Proposal of the criteria of 30 minute delay time to to take an action;

- Necessity to rely on instrument indication but with sufficient redundant and diverse instrumentation to be able to make cross-checking;

- Sufficient competence and training of the operators.

(3) Organizational aspects:

- The great importance of collecting, analyzing rapidly and exchanging information on incidents was emphasized; a project is under development in the frame of CSNI;

- Resident inspector: could be useful but may give rise to practical (human relations) problems;

- To be efficient, safety analysis must be integrated and not only made per system or project stage;

- Safety is not only a question of regulation but must be a preoccupation of all contributors to a project: designers, builders, operators.

The TMI accident highlighted a number of difficulties and weaknesses in safety organization and analysis; in this sense it was very useful. All the lessons to be learned have certainly not yet been extracted and doubtless we will have other discussions on this subject in the future.

We had no time to discuss some items such as emergency preparedness which are also of great interest.

SESSION VI

Mr. F. Cogné

La session VI était consacrée aux méthodes techniques d'examen et d'évaluation; elle a comporté trois sujet distincts :

- l'évaluation des risques d'accidents dans les centrales nucléaires allemandes effectuée par des études probabilistes de type Rasmussen ;
- le support technique via des programmes de recherche pour l'évaluation de sûreté réglementaire ;
- l'évaluation de la contamination et des conséquences radiologiques en fonctionnement normal ou après accident.

En reprenant successivement ces sujets :

(1) Dans le contexte allemand il a été jugé nécessaire de reprendre des études probabilistes de type Rasmussen (WASH 1400) pour tenir compte (a) de la conception particulière des centrales allemandes, et (b) des différences dans les sites et les densités de population entre l'Allemagne et les USA.

Voici un très court résumé du texte de la communication :

- la probabilité de fusion de coeur (4.10^{-5}/an en valeur médiane) est voisine de celle du rapport WASH 1400, les événements initiateurs contribuant le plus à cette probabilité étant aussi le petit LOCA avec erreur humaine et les transitoires ;
- l'évolution dans le temps de l'accident de fusion du coeur diffère d'avec WASH 1400 ; en général la défaillance de l'enceinte de confinement survient après 24 h environ suite à une lente montée en pression ;
- les conséquences sur le plan sanitaire à court et à long

terme sont voisines de celles du rapport WASH 1400.

La question a été posée de savoir pourquoi les résultats obtenus étaient proches de ceux du rapport WASH 1400, alors que les caractéristiques des sites sont très différentes.

L'utilisation pratique des méthodes probabilistes a aussi été évoquée.

La deuxième sujet abordé a été l'analyse technique de certains problèmes de sûreté rencontrés dans l'évaluation réglementaire de la sûreté. Il s'agit ici de programmes d'études et de recherche en support direct de l'évaluation réglementaire, ce qui est un aspect particulier des liens entre recherche et évaluation réglementaire traités à la session VII. Ces études et recherches sont nées de besoins précis rencontrés dans l'évaluation réglementaire ; en conséquence l'absence de tels moyens de recherche est un handicap sérieux pour les pays qui en sont démunis et l'on retrouve les problèmes évoqués dans d'autres sessions sur les difficultés et les limites de l'évaluation réglementaire dans certains pays. Comme les préoccupations et la nécessité de la sûreté existent dans tous les pays de manière semblable, l'idée a été avancée que les résultats des programmes en support direct de l'évaluation réglementaire devraient être mis sans restriction à la disposition de tous les pays intéressés.

Le troisième sujet, abordé déjà sous l'angle probabiliste dans une des communications précédentes, est celui de la contamination et des conséquences radiologiques en fonctionnement normal et en cas d'accident. Différents moyens de calcul ont été exposés. Dans la discussion la notion de coût/bénéfice et par là même la valeur à accorder à la vie humaine - tant pour le public que pour les travailleurs - ont été évoquées.

SESSION VII

Mr. S. Israel

The purpose of safety research varies among the different countries. In the F.R. of Germany, it is not the goal of the safety research programme to verify the conditions of the licensing processes. None of the safety research programmes are preconditions for licensing today. The basic structure in the FRG separates safety research and safety review. The exchange of information between the authorities and their commissions and institutions make sure,on the other hand, that the licensing requirements recognise the state-of-the-art as well as safety research recognises the needs and future wishes of the safety review processes.

In the US, the safety research programme is oriented specifically to the regulatory activities,so the research programme has been meeting its objective.

Other countries, such as the UK and France, have extensive research programmes related to their own national needs.

In general, the safety research programmes are oriented towards solving specific technical aspects and ensuring the safety margins, the applicability of standards and codes, and the protection of the population.

There is a need for the regulatory body to be directly involved in research activities if it is to effectively fulfil its statutory obligations to safety. This implies some ability to allocate and control these research activities.

The type of research activities engaged in by the regulatory body should be confined to those which enable it to perform its statutory duties in an effective manner. Large scale programmes should be left for the industry to sponsor. It is also undesirable for the regulatory body itself to carry the major responsibility for research, since this clearly lies with the manufacturer.

We should avoid redundant safety research programmes in different countries. There is a strong need for more international collaboration in this area in order that resources can be pooled and the information relevant to safety thereby increased. It will be useful in this regard that CSNI sponsor international meetings on research. One example of this is the Workshop on Consequence Modelling, scheduled for next year at Oak Ridge, USA.

At the opening of the meeting, Mr. Giuliani noted in his overview of the licensing process in Member countries, that the use of event trees and fault trees were starting to be used to review and assess plant safety along with standard deterministic criteria. These probability techniques had their genesis in research programmes and are now being integrated slowly into the regulatory process. The Lewis Report and Kemeny Report both support the use of these analyses as a means for assessing design reviews. This evolutionary process is an excellent example of the beneficial effects of research on the regulatory process. This does not mean, however, that research efforts in the application of probabilistic techniques should be curtailed, but rather additional effort should be considered.

An important point was raised during the discussion of research and regulatory review: What is the lesson from TMI-2 that should be addressed by future research programmes? One delegate suggested that the containment should be reconsidered because it represents the last barrier for protecting the public; while other members thought that additional probabilistic studies needed to be performed to identify the limiting scenarios, new assessments of fuel clad interaction for conditions between LOCA and core melt are needed, and re-emphasis on accident prevention is also needed. Since all of the speakers stressed the need for interaction between the regulators and research people to initiate research programmes, perhaps this is a suitable topic for future actions.

SESSION VIII

Mr. T. Eurola

The session dealt with five separate topics, namely:

- assessment and inspection interphases;
- backfitting;
- decommissioning;
- site evaluation;
- regulatory requirements on spent fuel.

1. Assessment and Inspection Interphases

This particular aspect comes up in a distinct manner when the regulatory body has set up a control mechanism, where the design review and inspection activities are performed by separate sections in the organisation. The interphase between two activities can also become problematic if the design information for some reason is not made available to the regulatory body until at the stage of manufacture or even later, when the final product has to be inspected.

In the case that the two activities are separated, there has to be a well established system within which the proper communication between the two can be carried out.

In the paper by Mr. K. Santoma, an analysis mechanism is proposed of the different activities involved in the design and construction of a nuclear plant, with due reference to the role of the parties involved. Based on this analysis, the regulatory body can arrive at an improved efficiency of its design assessment and inspection efforts.

As an illustrative example, the division of duties of the Evaluation and Inspection Sections of JEN is outlined for the activities connected with the design and construction of a mechanical system.

One of the important aspects pointed out in the paper is the due consideration of the fact that both design and construction constitute a continuous process where the interphase between the regulatory review and inspections has to be carefully evaluated.

2. Backfitting

Because of both the evolutionary nature of licensing requirements and technology developments, the operating nuclear plants include a broad spectrum of the design features. Thus there is an evident pressure on backfitting measures.

The programmes that are already in place for backfitting and the systematic evaluation of the operating plants will, expectedly, receive additional emphasis in the future.

The paper by Mr. D.M. Crutchfield described the overall backfitting scene in the US. It reviews firstly:

- methods of backfitting;
- assessments for backfitting.

Then it describes in detail the two important programmes related to backfitting, namely:

- systematic evaluation programme;
- the short term safety-related recommendations of the TMI Task Force.

The systematic evaluation programme was initiated about 1½ years ago with the aim of assessing the safety adequacy of 11 of the older operating plants.

Experience has shown that a need exists and is recognised to maintain a systematic evaluation programme for operating plants. Although the structure and format of such a programme may undergo changes over the next few years, the goal of improving safety at operating nuclear facilities will not diminish.

3. Decommissioning Licensing Procedure

After a general outline of the problem area the paper by Mr. Perelló goes briefly through the most frequent methods of shutting down nuclear reactors.

Four main decommissioning methods for major facilities are given.

The main point of the paper is the suggestion that a new chapter be included in the PSAR, namely, "Preliminary Programme of the Decommissioning of a Nuclear Power Plant". It is recalled in this connection how a new chapter of PSAR on quality assurance came into being rather recently.

Three methods of decommissioning are proposed to be considered from the regulatory point of view:

- mothballing;
- entombing;
- dismantling.

In conclusion, it is pointed out that:

- decommissioning is a necessary action;
- safety has to be considered to the very end of the plant's existence;
- the decommissioning has a very direct influence also on the site considerations and financing the plant.

I would propose that the CSNI consider arranging an opinion poll in OECD Member countries concerning the suggestion given in the concluding part of the paper.

Panel Discussion

P. Giuliani (Italy)

Before giving the floor to Mr. Stadie of the NEA, I would like to make a small comment myself on this meeting. As a matter of fact, I am sorry this comment is a criticism to myself, as Chairman of the Working Group which organized this meeting. We had here quite a number of excellent papers on different subjects, but the mere fact that we listen to so many good papers gives me the idea that possibly we had too broad a scope for this meeting, because if we had kept the scope a bit more narrow we would have got the same good quality of papers but on some more specific subjects. Actually we call it the old scope of the licensing review and this is a tremendously broad scope. As you realize it's difficult to cover in three days. So with this comment I pass the floor to Mr. Stadie of the OECD NEA.

K. Stadie (NEA)

Thank you, Mr. Chairman. Before starting I would first like to clarify a point; You, Mr. Chairman, have introduced me, as representative of NEA; I consider myself on this part of the panel as a simple observer, participating in the meeting, and I would like to give you my personal views of this meeting. As a matter of fact I would like to give you some of the observations and some recommendations which I see come out of this meeting, and then I would like to continue where you started and give a sort of maneouvre critique to see where we can do better in the future. I have not been able to relate my comments to any common denominator other than perhaps, that they somehow all relate to TMI - TMI which I'm sure is going to haunt us all in the years to come -and of course they relate to international cooperation, but that, coming from me, would not be much of a surprise to you. So let me just give you some reflections and recommendations.

I'm starting out by recalling what Commissioner Kennedy said on the first morning. He said that it is imperative now to have as broad as possible an exchange of information on operating experience of power reactors. As I mentioned to you on the first morning, we are trying to set up such a system. The value of assessing incidents in nuclear power in international cooperation has been recognized very early. As a matter of fact, the first meeting of

CREST in 1966, I think Dr. Alonso will remember, already examined incidents in reactors, and tried to draw lessons from these incidents. CSNI has continued this practice over the years, and it has become more and more cumbersome a problem as the number of reactors increase, and thus the number of incidents.

At the last meeting of CSNI the United States proposed that we should systematize more this exchange, and as a result of this CSNI set up a working group, as a matter of fact under the chairmanship of Mr. van Daatselaar on my right. This working group has developed criteria for reporting these events to a central mechanism for distribution to all member countries. From the technical point of view the work of this working group was very successful and CSNI will discuss next week whether or not to support it. There is no doubt in my mind that this will be very strongly endorsed, but unfortunately this will not be sufficient; there are a number of contries within the OECD family which have difficulties to transmit this type of information to an international organization or across national borders. So here is my plea: I would wish that all of you who are interested in such an exchange, to do your utmost to overcomo these administrative burdens. Our plan is that after this CSNI meeting we will approach the various countries who have difficulty in this respect to tell us about the administrative hurdles, and we will then call a meeting of government representatives and see whether we can sign a sort of international agreement which makes this exchange possible.

A second proposal which came out of a meeting of the Bureau of CSNI, which consists of Professor Birkhofer, Mr. Gausden, and Mr. Levine, and in which Mr. Tanguy and Mr. Sato from Japan also participated, concerns a suggestion that everybody has another look at the LER reports and see whether he finds something that one could have seen in the Rancho Seco incident, and if so communicate it to us for distribution to Member countries.

A third proposal which also came out of the enlarged Bureau meeting concerns the preparation of a publication summarizing all new and reoriented safety research being undertaken as result of TMI.

Then I have another proposal which relates somewhat more to the discussions you have here during this week; that concerns the questions of how are we are going to assess Class 9 type accidents in the regulatory process in the future. This must be a question you are all asking yourselves and I feel that it would be useful that the Subcommittee on Licensing should meet soon to discuss this question informally. It is evident that the decision whether or not there is a need to change current practice in licensing is a national decision. But I think before any national decision is taken, it would be extremely useful for those responsible in this field to exchange views.

In the second part, as I mentioned, I would like to make a short maneouvre critique and I am criticizing myself here, because it is after all NEA and CSNI who sponsored this meeting. I was very impressed with the meeting two years ago on inspection practices. It focused on a specific aspect in the overall regulatory picture, and I think everybody who has been there agrees with me that it was extremely useful. This meeting, as your Chairman already said, was probably too broad to be covered in a three-day meeting. There were certainly excellent contributions and at times very interesting discussions, that it was a very broad meeting, you can see, that over the three days participation continuously changed, it looked almost like a miniscale Geneva Conference in the regulatory field. Therefore my suggestion that in future one should perhaps, in order to improve the effectiveness of the international exchange, cover a more limited topic and allow more time for discussion, rather than for presentation, which is necessary when the topic is broad. Now if I be asked to suggest some topics to take away from this meeting, I would think an obvious one is operator training and qualification. I think everybody is asking himself now, are we doing the right thing, are we training our people to run these reactor in the right way, what should be their qualifications, what should be the limit of their responsibility? etc. Another question, which I think was the topic of a special session during this meeting but which was not really properly treated, was the relationship between safety research and regulatory review. We had a number of interesting discussions on specific safety research questions, but really no discussion evolved or developed on the interlink between the two. There were some thought provoking comments and suggestions, I think, in Mr. Levine's paper, but they were not followed up. So I think this may be a very interesting point to take up sooner or later.

And finally we had the recommendation this morning to discuss or do some work on decommissioning. As I already mentioned this morning, I felt that here the emphasis is more on radiation protection and waste management. At the same time I do realize that there are safety implications and since the Subcommittee on Licensing brings together the regulatory bodies in the OECD area, it seems appropriate to pass the Spanish proposal to the Committee. And if as a result of that some specific activity should evolve, one would have to look into the question of either dealing or examining this question jointly with the Committee on Radiation Protection and Public Health, or Waste Management Committee, or defer it entirely to these Committees.

Mr. Chairman, I think this is all I would like to say for now, but I would certainly welcome to have from the other members of the panel or from the floor, reflection on some of the recomendations and the criticisms I've tried to make here.

P. Giuliani (Italy)

If any of the panel members has anything to comment on Mr. Stadie's proposal or ideas?

J. van Daatselaar (Netherlands)

I'm sorry, it was not my intention to take the floor first, but Mr. Stadie mentioned the assistance for exchange of information and I think it certainly will be an important thing to do, but if you look at the criteria for sending this kind of information, it's mainly focused on incidents or near incidents. And it's also been said that the Enlarged Bureau will go more into detail into the licensee events reports. But all the licensee events reports reports in the United States did not prevent the TMI accident to happen, and I think the main reason is that much attention is always given to what they call 'structure systems and components important to safety'. But if we look at the power plant, then your first line of defence is the control system to keep all the processed parameters within the operating limits. But in the second phase you are relying on your safety systems to prevent the safety limits to be exceeded. So I think that maybe in the future we should give more attention to the first line of defence, and maybe in discussion in next year's meeting, when we discuss this system of exchange of information, we could give some attention to this point. I have to admit I did not do it in our meeting on the setup of the exchange of information system.

That was what I would like to comment at this point.

K. Stadie (NEA)

Yes, I did not, Mr. Chairman, go into the details of the criteria for reporting, which are in the view of some, I think in the view of many, too restrictive, particularly as concerns the US Department of Energy. At DOE I was criticized for NEA trying to support such an unambitious international exercise. Of course, this is the sort of criticism I like best. But I shall caution against a too zestful approach next week at the CSNI meeting, because knowing the difficulties of setting up any kind of exchange among 24 countries, I believe that the only way to get started is with a very humble system and once we have proven that we can handle it -and still I have to recall to you that about 50% of the member countries need some legal provisions to make this reporting system possible- expand the system. So therefore while I agree with what Mr. van Daatselaar said, I think it is appropriate that we start with the LER report as defined by the NRC, in the first instance.

G. Tenaglia (Italy)

I would like to raise some question about what Mr. Giuliani and Mr. Stadie said about the broad scope of this meeting. Yes, it's true, the scope was too broad, but it seems to me something was missing. I mean, just to clarify the limits and the the scope of the regulatory review. We had some speech, some statement made by Mr. Kennedy, and he said there is responsibility also belonging to the management of the operating plant. That is true, but if we don't try to define completely - I wouldn't say clear-cut, that is impossible - the responsibility of the management of the plant, of the designer, of the regulatory review, I think we are just bringing the responsibility to other sides, and then not clarifying what are the limitations in the regulatory review. It seems to me that this is completely necessary. We can't do everything, there are limits, and we have to stress this point, and then give the responsibility to other sectors of the nuclear area, and this must be clear, otherwise we are just taking point by point - I mean operator training, personnel, okay -but what are we expecting from them, what's their duty, how much can we rely on them, what's their responsibility, so we can know exactly what is our duty in front of them. It seems to me that this kind of clarifications, like those concerning regulations and standards, when we have them it seems that we have solved the problem and I say no, we din't solve the problem. Here, the same, There is a regulatory review, it seems we have solved the problem of the safety of the nuclear power plant. This is not true. So we must stress this point and clarify the border, the limitation that we have. This is an important point, maybe this is a suggestion for a meeting, an open meeting, really - open in the sense that it must be a clear speech, and I heard this many times during this meeting. There are limitations of personnel, there are limitations in information and so on. Yes, in front of which kind of duty a regulatory review? This is my suggestion to think about, if there is not a meeting, if you think it's too much, let's try at least to clarify it to the men, to people involved in the regulatory review. Thank you.

P. Giuliani (Italy)
Thank you, Mr. Tenaglia. You've certainly provided us with a lot of food for future thought.

A.Acha (Spain)
It is possible that my outlook, my point of view, could be partial. I don't think so. I've noticed that throughout the meeting very little attention has been given to the problem of siting, either site survey or site qualification. I believe that touching on this matter, which is so difficult since one must study it case by case, extrapolating the experiences of other countries, it would be very useful to provoke, to prepare a meeting. Thank you.

P. Giuliani (Italy)
Well, as a matter of fact, you know that you find me very open on this, because it's my favourite field. And I certainly agree that site meetings, on siting aspects and siting problems, should be held. I'm sure that many of them are being held already. Possibly what you are suggesting is something which is oriented towards licensing problems in siting. This could be one of the aspects. As a matter of fact, we tried to cover this aspect when we prepared our questionnaire one year ago. But we didn't get much response on this for a number of reasons, one of the reasons being that possibly some people felt that they already had enough on siting. But I agree that seeing siting from the licensing viewpoint is possibly interesting, and comparison of different siting practices is even more interesting, especially for countries which are in a less advanced stage of development of their nuclear power programme. And of course in Italy we are very much interested in this area and I know you are in Spain and Portugal too, and so many other countries

are interested in this. Well, I throw the suggestion to Klaus Stadie. Do you want the floor?

K. Stadie (CEA)

Yes. Thank you, Mr. Chairman. I have always been a sceptic regarding international cooperation on the question of siting, and I think I could simply explain that by saying that Japan, Canada and Finland are OECD members. I think this morning we saw a good example of where collaboration outside OECD, but maybe nurtured by OECD, could take place; this was the question of seismic problems relating to the Mediterranean countries.

There seems to be a good reason for these countries with similar geology to get together and try to develop guidelines or carry out research in this field. But I find it very difficult to defend vis-â-vis CSNI a proposal which is obviously of interest, to a few member countries only. Some countries have enormous siting problems -Belgium, for instance, with a big population density; we heard from Finland earlier in the meeting that their siting problem was not really a very considerable one, so I would certainly caution against going too deep into this field. Evidently there are certain aspects of siting which from the licensing point of view are of interest to most member countries.

R. Gausden (UK)

I think Mr. Stadie is touching on perhaps what is a much wider problem, and I would welcome his comments. We have in the field of nuclear safety generally two organizations who are to some extent competing with each other to take the floor. I am referring of course to CSNI OECD on the one hand, and the IAEA on the other. Now I know that Mr. Stadie will say that there is cooperation between the two organizations, and there is some coordination of effort. And to some extent this is true, but in the long run it is you here and people like you who have to service both organizations, and this is a problem, and perhaps Mr. Stadie might like to say how we get over this.

K. Stadie (NEA)

Thank you, Mr. Chairman. I think it goes a bit beyond the scope of this meeting but I am of course quite happy to answer this. I would start by just underlining what Mr. Gausden has said: there is cooperation, there is coordination. As a matter of fact, if you look at the entire field of safety regulation, there is more coordination in the field of safety than there is, for instance, in the field of waste management and radiaction protection. Until recently, until TMI, the line was very clear, namely that with CSNI, arising from the historical development of CREST, our aim was to facilitate cooperation on safety research in the broadest sense possible, while IAEA would concentrate on codifying the composite experience in safety. I say this was until TMI because, as I mentioned the first morning, there are certain initiatives to broaden the role of the IAEA, and what I have said has worked in the past may prove difficult in the future. But also on the first morning -I think-. I gave a number of reasons why we believe that the type of activity which CSNI is undertaking is impossible to be urdertaken by IAEA. This relates to the fact that safety research as such does not exist in the Eastern countries, that safety research as such is not going to be sponsored by the Third World countries; it is therefore the responsibility, so to speak, of the OECD countries to carry the burden in this field, perhaps with us helping to coordinate it.

It is slightly different in the field of licensing. In the field of licensing we have, when CREST was transformed into CSNI, provided in the Subcommittee on Licensing a forum for the exchange of information between licensing authorities. Under the chairmanship of Mr. Gausden the subcommittee on Licensing began by discussing a number of sensitive issues off record, and it then, a few years ago, became somewhat move operative, in a similar fashion as the main CSNI Committeee. One of the outcomes of this type of cooperation is the specialist meeting we are conducting here.

I have no crystal ball and I cannot of course foresee what is going to happen, but I feel if the Subcommittee on Licensing would develop along the two lines which I tried to indicate before it has the best chance of being a useful tool to our member countries. Namely, to go back to discuss sensitive issues such as what do we do with Class 9 accidents in the future, and to sponsor from time to time restricted meeting on specific topics; and in spite of what Mr. Tenaglia said I would insist that a meeting of the kind we had here was too broad. It would have been more useful maybe to have two or three separate meetings spread over some time on some of these topics which were forming part of this meeting. I would think therefore that in these two ways, namely, discussion of sensitive points in a plenary meeting, or even in special meetings if this proves necessary, and a modest operative programme of this kind but again with more specific topics, would be the best assurance that there is a minimum of overlap with IAEA. Thank, Mr. Chairman.

P. Giuliani (Italy)
Thank you, Mr. Stadie. Well, you gave us a good picture of the OECD activities in general and related to this type of meeting. Is there somebody who wants to ask questions on any topic which was touched during the three days, something still hanging in the air?

J.L. Butragueño (Spain)
Thank you, Mr. Chairman. I wanted to call the Committee's attention to something that, I think, has been spoken about a great deal these days, and which I consider one of the most important lessons learned after the TMI accident. I think we have all learned that the human factor has been one of the most important causes of this accident. It seems to me that all the recommendations put forward to prevent an accident like this in future focused on matters which I believe are partial, and I should like us to reconsider our point of view a bit. I mean to say that a lot of stress has been placed on intensifying training programmes, improving procedures and things of that sort. I would propose that more attention be given to the most important lessons learned at TMI, and that is that the human factor is an essential component of any safety system. The human factor should play its part as a safety-related element, not as something that is convenient or cheaper than something else - it's a safety-related matter. In this connection I think the Committee should give much attention to the manner in which the human participation is integrated as a whole within a single unit: the nuclear plant in operation. Thank you.

P. Giuliani (Italy)
Well thank you. I think we all agree that the human element is very important in nuclear safety and this was one of the lessons which came out of Three Mile Island. Of course it was known before, but I think that the human element, just because it's a human element is very difficult to quantify and to express in neat numbers. It's very difficult to use a precise relationship when you are dealing with an operator which may not be very bright or not very well trained, and you can do something about that. But he can be upset for personal reasons and so his efficiency may be lower than normal, and we are humans.
Now there is another point. I was involved for many years in licensing of operators for nuclear power plant and research reactors, when we were young, and one of the most common complaints of operators was that their job was terribly boring.

And of course it is boring and we are all thankful it is boring, because Three Mile Island accidents are very exciting but they should not happen every second day. Otherwise we would be out of business. The real thing is that it is very difficult to provide some entertainment for the operator during the long hours during which the plant is on line because well, Mr. Israel quoted a number of happenings which can brighten up the life of the operator and may happen, let's say, twenty times a year. That's, well, a good number, maybe it depends on the degree of entertainment the operator wants to get out of them. But really that's a point out of which it is difficult to come because the station is supposed to work at full power for long periods of time. It was said that, for instance, one of the reasons why, naval propulsion plants were better as far as the operator was concerned, was that they had a more varied operation. I doubt that really but, it may be- I don't know much about that area.

Another point which relates to what you say about the human element being important is the human engineering of the control room. Control rooms are dull places after a while. And again, much can't be done in that area but as we were saying some time ago with Mr. Israel again, the real problem with nuclear power plants is that they are run and operated and designed by engineers and engineers are very dull people, as I am, and everybody else who is an engineer knows. And so we tend to design things which serve a purpose, and then, that's it. So it's, you know - geologists are much brighter. So it's a very complicated area.

I think something can be done but within reasonable limits. Of course if you look in the nuclear safety literature you'll find articles and things speaking about reactor operators and the human element in the safety analysis. But it's like reading articles on nuclear safety written in the fifties. We are still in the beginning of this and if I remember correctly - I may be grossly mistaken - but I have a feeling that the reliability number, if you want to call it a number, for an operator was about ten to the minus one, which is not much, really, but that's it.

Another point: power station's control rooms are an outcome of the control rooms of research reactors on one side and oil fired power stations on the other, or just any sort of automated industry, and you know that research reactors usually are a source of radiation, neutrons or gamma or whatever, and so the control room is optimized toward a certain aim. On the other hand, a power station which is coal-fired or oil-fired is strictly for winding out megawatts hours. And so it's two types of dull things. Coupled together they make a dull control room. And that's it.

K. Stadie (NEA)

Thank you, Mr. Chairman. Just to repond to the intervention from the floor concerning human errors,CSNI has a working group on human error data and assessment wihich has, started out with a very humble task just collecting relevant data and to develop models for analyzing human error. Now I understand this is an extremely difficult task and I don't need to go into detail, you can very well imagine that very little has been done here collecting data under normal working conditions, and it's much more difficult to determine behaviour in a stress situation because other factors come in which are yet more difficult to measure. But there are some studies apparently which are being developed in order to quantify human error in this field. Thank you.

S. Israel (US)

Yes, I guess I would like to make a few comments dealing with the humam element. As Dr. Heuser pointed out in his risk study, in the small break LOCA, sixty percent of the unreliability was attributed to the human element, and I believe he used an unreliability factor for the human element that, I think, was two times ten to the minus one. I look at it a bit differently. I know that people are trying to collect the data and come up with a number as far as what the unreliability of the operator is, and that's like closing the barn door after the horse has escaped. I think we should look at it in a more positive vein and try to come up with a conceptual model as to what we really expect of the operator in the control room, and how best we can improve or enhance his actions so that we have the maximum of benefit, rather than collecting data on previous experience that maybe are not very useful in regard to risk assessment, because it may be very unhappy data.

C. Pérez del Moral (Spain)

I should like to enlarge on the idea that has been brought up of having a session devoted entirely to this subject, or using any other means to take a closer look and see what are the limits and responsibilities of a safety regulatory body, as Mr. Tenaglia mentioned. I think we are all agreed that a safety regulatory body has no need to redesign a nuclear plant, but there should be a middle point between the redesigning of a plant and the mere pursuit of quality assurance. It should therefore be important to decide just where this middle point lies in the safety regulatory body. I can think of a few things that must be decided upon - for example, in which areas or on which points the concept of the reference plant should be applied, what an independent calculation should focus on - I mean, I think an independent calculation can't be similar to one done by a designer or vendor; perhaps one should see what are the most meaningful parameters, for example in the development of a transient, and thus determine what are the most urgent sensibility studies in relation with this transient or accident. Some other things occurred to me but I can't recall them now.

P. Giuliani (Italy)

Mrs. Pérez, I gather that you gave a number of possible subjects for future meetings or something like that.

C. Pérez del Moral (Spain)

Yes, that's what it's all about. The problem, as has been pointed out here, is that we seem not to agree, for example, on the number of persons needed for a good safety regulatory body bearing in mind the type of plants concerned rather than the number of plants. Because one can have many reactors but all of the same type and on the same site. In this case especially the number of persons is not simply related to the number of units. But I don't think we will agree on this point if there isn't a clearer definition of what are the responsbilities of a regulatory body and what we think its duties should be.

P. Giuliani (Italy)

Thank you very much. Actually, Mr. Acha will understand why I was smiling. Well, the reason I was smiling is that last week I was in Argonne and I gave a three-hour lecture on exactly the same topic and Mr. Acha was present. So you can start asking him.

But I agree that these are all very basic topics and they could and should be dealt with in a bit more detail. If I may keep the floor for one more moment, you mentioned limits and responsibilities of a regulatory body and number of people you need for staffing a regulatory body in different situations and so on. Now I don't want to advertise for the International Atomic Energy Agency, but if you look at the guides in the NUSS Programme, that is, the five series of guides they have produced, there is the series on Governmental Organization which is really very good, I must say, and they have a lot of useful things about this specific topic, that is, limits of the applicant, number of staff, type of training and this sort of thing. Of course with those documents in the NUSS Programme, they tend to be general, because they represent a kind of average. But they have very good basic ideas.

Does anybody want to comment on that or on any other things discussed?

P. Lebouleux (France)

Merci, monsieur le Président. Je veus faire un petit commentaire à la suite de cette question. Je crois comme même que les choses sont très claires. Les premiers qui font la sûreté des installations, ce sont bien les constructeurs et les exploitants. Nous autres, organismes de sûreté, nous ne devons pas avoir la prétension de refaire toutes les études fondamentales et de base, qui ont été faites pour amener les systèmes des reacteurs actuels à fournir de l'énergie éléctrique ou d'autre source d'énergie. Je crois que nous sommes là essentiellement pour vérifier que des études suffisants ont été faites, pour s'assurer que toutes les mesures, enfin, que les demonstrations qui nous sont faites par les gens qui demandaient le permis de construire, d'exploiter, etc., que toutes ces démonstrations sont satisfaisantes et sont bonnes. Et c'est pourquoi au cours des différentes présentations que nous avons faites au cours de cette petite réunion, nous avons justement appuyer le fait que l'organisme réglementaire devait être, ainsi que ses appuis techniques bien entendu - devaient être suffisamment, néanmoins, armés pour pouvoir, indépendemment, mener certaines études parallèlement à ce qui est fait par les constructeurs et par les exploitants. Mais je crois que vous avez parfaitement raison. Il n'est pas pensable de se mettre à la place de gens qui ont construit ou réalisé le réacteur dans tous les domaines de leurs activités. Mais c'est le domaine important pour la Sûreté. Il faut que nous soyons en mesure de nous former par nous même notre jugement suffisamment indépendent.

R. Gausden (UK)

I would like to very much associate myself with the point that has just been raised. It worried me very much when the statement was made down here that the safety body should be concerned with the design or redesign. I don't think the safety man is concerned at all with that function.

G. Tenaglia (Italy)

Thank you, Mr. Chairman. I think I would share the opinion of Mr. Lebouleux, but I think there is a confusion around the world because if something happened, the regulatory body is the first involved by the political or anyhow by the population, because it is, we think - or they think - primarily responsible for safety. So it seems to me that we agree but not completely with the people - this clear statement isn't clear. And then we have also to clarify, not to discover, after Three Mile Island, maybe that the management of the nuclear power plant has its responsibility. Because we have to remember very often this part - it seems to me that things are not completely clear. So that was my suggestion, trying to clarify a little more or to stress this point which are the relative responsibilities. Thanks.

K. Stadie (NEA)

Thank you, Mr. Chairman. May I also say something to this question, the matter of responsibility. Nobody can quarrel with Mr. Gausden that it cannot be the responsibility of the regulatory authority to design a reactor, maybe even operate a reactor. On the other hand, I note that at least in the translation, Mr. Lebouleux said that the regulator should be 'sufficiently armed'. What does it mean exactly?. Many people cannot do much with such unclear advice, and I therefore have a lot of sympathy with what Mr. Tenaglia says. Thank you.

C. Pérez del Moral (Spain)

Let me just say, after Mr. Stadie's clarification, that I think I was misunderstood. From Mr. Gausden's reply I believe I was understood to say that a safety body had to redesign everything. My proposal was not that, but rather that the responsibility of the body be determined, as Mr. Stadie has made clear.

J.M. Izquierdo (Spain)

The question is directed to Mr. Israel, because he is the only person from the NRC there, and the question is very concrete and goes into this subject. After Three Mile Island it was clear that NRC has been severly criticized. On the other hand relative responsibilities between NRC and the industry are not clear even in America. You have none the less around 2700 people. What's the feeling of NRC now? You need 5,000? You need to transmit the responsibilities to the industry; or do you need to clarify to the people that you are only backing up? What's the feeling now? It's a pure question.

S. Israel (US)
The position of the Commission vis-à-vis the industry or the utilities has really changed. The utilities are responsible for operating the plants under normal operating conditions, on the transients , and under post-accident conditions as Three Mile Island was in. I don't think it's a matter of number of people, I think it's a matter of what our charter is, and what it is that we are affirming to the public. And basically we are affirming to the public that there is reasonable assurance that the plant can be operated, and yet protect their health and safety. Now the definition of what protects their health and safety is all codified in our Code of Federal Regulations 10 CRF 50. Three Mile Island of course has brought into question whether those codes are sufficient as far as the public may be concerned in providing reasonable assurance. I mentioned yesterday that - I think I mentioned yesterday - that part of the lessons learned - the final report NUREG 0585 - one of the recommendations is our developing a clear-cut policy on what constitutes safety, what is safe enough. In the past obviously we've been dealing with single failures and that type of limited approach, and based on that we've said, yes, we have reasonable assurance that the plants are safe to operate. The public's perception, obviously enhanced by the Kemeny Report, which indicated that we should be looking at multiple failure and all sorts of other things, may cause us to reexamine what safe is safe enough. I'm not sure that it will be done by the Commission itself, but it'll be done by Congress, probably, or some combination of both, with some input from the people. Because we are constantly engaged in this dialogue with the staff, it says, the plant is reasonably safe because it meets the federal regulations,and the public questions whether that is sufficient protection for them. And Three Mile Island obviously has brought this all into question and it will undoubtedly be reexamined over the ensuing time.

J. McLeod (UK)
Just a point, since Mr. Israel when he was summing up the session quoted a bit that I said. One bit that I didn't say but which is in the paper, which I read now: 'It is also undesirable for regulatory body itself to carry the major part of the responsibility for research, since this responsibility must clearly lie with the manufacturers, although some will be independent in origin, though an area of overlap between research activity is inevitable, the research sponsored by the regulatory body should be essentially exploratory or funda-

mental in nature by virtue of the above, because its resources are limited.'

I didn't say that in the verbal address, I left it out, but as far as research is concerned, I've got a clear demarcation of responsibility.

P. Giuliani (Italy)

Thank you, Mr. McLeod. Well, I don't know, if there are no more interventions, we might, I might try to summarize in a few minutes what we said.

Quite a few interesting items came out. Some of them had been already mentioned during the meeting. The human element seems to be, and actually is, a very important concern with many of us here. Some more subjects for future meetings have been proposed, and the chairmen of the different sessions summarized their own session, so there is not much left for me to do but to thank on behalf - I am the last chairman here, so I may take the liberty on behalf of all of us who came from abroad to thank very much the Junta de Energía Nuclear, its Chairman, Mr. Olivares, Mr. Pascual, Mr. Alonso and all other Spanish friends, for the delightful hospitality - Mr. Reig, of course, and Mr. Trueba - I cannot count them all. And I'll call on Mr. Alonso, as Chairman of the meeting, for final remarks. Thank you very much.

CLOSURE OF THE MEETING

Closing Authorities: A. Alonso
R. Gausden
K.B. Stadie
P. Giuliani
P. Trueba

Concluding Remarks

A. Alonso (Spain), Meeting Chairman

Mr. Stadie, CSNI Secretary; Mr. Gausden, Chairman of the Subcommittee; Mr. Giuliani, Chairman of the Working Group; distinguished representatives from the different countries and International Organizations; ladies and gentlemen:

It is now time for me to deliver my Concluding Remarks, the difficulties of which task have been eased by the ability and dedication of the Chairmen and Scientific Secretaries of the different Sessions, and by the undoubted experience of the President of the last Session and Chairman of the Working Group, Mr. Giuliani.

The report on the Regulatory Review in the Licensing Process in the NEA Member Countries, which was prepared by Mr. Giuliani following a request from the Subcommittee, has provided interesting information, mainly to representatives from those countries with limited manpower resources. The paper, I am sure, has generated, and will go on generating, ideas on how to cope with the problems posed to those of us with the task of assessing and reviewing nuclear power plants from the regulatory side. In my opinion, this type of work should continue within the framework of the Subcommittee.

It became very clear that the most advanced countries have progressed farther than most, for obvious reasons, in establishing their assessment and review organizations, as well as in developing and implementing codes, standards and guides.

The not so developed countries are also trying hard to establish their own regulatory bodies following the principles of independence, competence and authority which come clearly from the most advanced countries. While independence and authority are easily implemented by the appropriate laws, the development of competence still remains to be solved, as it was clear from the discussions. The solution to this problem is difficult to find for most countries. The Subcommittee on Licensing and other bodies can help, but the transfer of competence presents intrinsic difficulties, which can only be overcome with time and by participation in the acquisition of knowledge.

The meeting has shown that there are very few countries with complete sets of codes, standards and guides; there are also well developed countries without these complete sets, despite the fact that they manufacture standard nuclear power plants.

It can also be deduced that the less developed countries are engaged in the process of establishing codes, standards and guides, albeit of a more administrative rather than purely technical nature. Moreover, it was also clear that having a complete set of codes, standards and guides is not a sufficient guarantee of having a safe plant. Doubts were expressed about the fact that relying only on these documents would lead to complacency and, therefore, inefficiency. This corresponds to the well proven fact that over-regulation does not necessarily mean that the safest design is achieved. In safety, one has to be alert all the time and use common sense and engineering judgement.

Probabilistic vs. deterministic approaches to safety have been discussed at length during our Conference. Work in the Regulatory Organizations consists mainly in making a low risk activity out of an intrinsically very dangerous enterprise. Fission products accumulate in the reactor core and it is our duty to make sure that they do not escape from where they belong. Therefore, evaluation of risk is our business and it involves the calculation of accident probabilities. If we are not using probabilistic analysis it is because of our ignorance, not because we do not need this type of analysis, which is certainly at the very roots of the matter.

We must recognize, and this became very clear from the discussions, that we have made a long progress since the old Canadian and British studies on the probabilistic approach; the Rasmussen Report and the most recent German Risikostudie are important steps, which are clearly throwing light on research and on the licensing process. We will be moving closer and closer to the full use of the probabilistic approach.

The Conference has been exposed to the late developments resulting from the TMI-2 accident, not only in the U.S.A. but in other countries as well. The concerns of and actions taken by the countries vary with their industrial development. There have been reactions covering a wide spectrum, from very basic concerns, such as being able or not to cope with the circumstances of a serious accident, up to bringing into question some of the actions taken by the U.S. authorities.

The discussion on TMI-2 centered around the interphase between the brain of the man operating the machine and the machine itself. This is certainly a difficult problem simply because it will not be possible to turn a man into a machine - however well trained he may be - or viceversa, ie, it will not be possible to make a machine think, however complex it may be in design and build.

The Conference has also touched upon two points of interest, which are at the beginning and at the end of the life of the plant. I am referring to site analysis and decommissioning, problems which have been brought up by Spanish papers. Site analysis is of interest because there is no such thing as a reference site. The resources of the country have to be used in the analysis and very little outside help can really be obtained.

Decommissioning is another matter. It brings radiological protection problems, but safety has the main aim of preventing people from being exposed to radiation through sensible design and operation of the plants. If one thinks about decommissioning when designing and operating the plant, the radiation received by workers participating in the operation will also be reduced.

In a meeting like this it is not possible to avoid talking about the impact of nuclear technology on our society and the response of society to our activities and concerns. It is clear that nuclear technology is maturing and is becoming a fact in the normal lives of people. A solution to the problem of public acceptance rests, from the technical point of view, on demonstrating that nuclear power plants can be operated safely - the situation at TMI-2 is a drawback for this. From the administrative side, it must also be proved that the assessment, review and inspection bodies, both private and public, are independent, competent and have the authority to correct deviations from standard practices.

Madrid, November 9, 1979.

Closing Remarks

K.B. Stadie (NEA), CSNI Secretary

Dr. Alonso, Mr. Giuliani, Mr. Gausden, Mr. Trueba, ladies and gentlemen:

As I am the last speaker I have the privilege to thank on behalf of all of you, the Spanish Junta for having invited us to Spain, and in particular to thank them for their warm hospitality. I also should like to thank Dr. Alonso and his many colleagues for their efficient support of this meeting. It was almost flawless, except, as you remember, on the first morning with the breakdown of the electronics, which I think was an appropriate sign for a meeting on safety because it reminded us safety engineers that machines, and not only men, develop faults. I should like to thank the chairmen for conducting their sessions efficiently and working so hard to prepare the summaries for the Panel Session. In this respect I would like to thank the Scientific Secretaries. Finally I would like to thank the participants who actively participated in the discussions and faithfully filled out the yellow forms. And last not least I would like to thank the interpreters - which enabled us to communicate with each other so effectively.

I now wish you a safe return.

Closure Message

Tte. General Olivares

President of the Junta de Energía Nuclear

He seguido con mucho interés el desarrollo de esta reunión de trabajo sobre una materia de tanta importancia y trascendencia como es la evaluación de la seguridad nuclear en el proceso de autorizaciones de centrales nucleares. Espero que sus resultados sirvan para continuar la necesaria colaboración internacional en esas materias y que hayan contribuido al mejor conocimiento de los temas aquí tratados.

Todo ello servirá para continuar el esfuerzo y la atención que requiere la seguridad nuclear. Agradezco una vez más su esfuerzo y dedicación al mejor éxito de esta reunión. Espero que se lleven todos un grato recuerdo de los días que han estado con nosotros y lamento no poderme despedir personalmente de ustedes.

(English Translation)

I have followed with great interest the development of this meeting on a subject as important and far-reaching as that of regulatory review in the licensing process of nuclear power plants. I hope that its results will help to further international cooperation on this subject and contribute to a deeper knowledge of the matters we have dealt with here.

All of it will be most useful in keeping up the effort and the attention that nuclear safety requires. I want to thank you once more for your work and your devotion to the greater success of this meeting. I hope you will all take away a pleasant memory of the days you have spent with us and I regret my inability to bid you goodbye personally.

APPENDIX

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CSNI SPECIALIST MEETING ON
REGULATORY REVIEW IN THE LICENSING PROCESS

Madrid, 7th-9th November 1979

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