AAEC/RERB-79-3

REPORT ON OVERSEAS VISIT

JUNE-JULY 1979

by

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Issued: November 1979

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PART I

PURPOSE OF VISIT

The major objectives of the visit were as follows:

(i) to attend the NEA CSNI and the CSNI Sub-Committee on Licensing and Regulation special meetings to discuss the accident at the Three Mile Island Nuclear Power Station in the United States;

(ii) to visit nuclear energy agencies in France, the United Kingdom, Canada and United States for follow-up discussions on nuclear safety issues.

PART 1

SPECIAL MEETING OF THE CSNI - 27 AND 28 JUNE 1979

2. The special meeting had been called originally for May to discuss quantitative risk assessment. After the Three Mile Island accident the United States delegates would have been unable to attend, and it was subsequently decided to postpone the meeting and devote it primarily to discussion of the accident. There was nevertheless an opportunity for brief discussion of quantitative risk assessment under item 7 of the agenda.

Address by Mr. I.G.K. Williams, Director General of the NEA

3. Addressing the Committee, the Director General of the Nuclear Energy Agency said that the prime reason for arranging the special meeting had been to ensure that the lessons to be learned were applied as rapidly as possible in all contries concerned; these lessons were especially relevant because a light water reactor had been involved.

4. Many demands had been heard for assurances that nuclear power should be made "absolutely safe"; in responding, the nuclear community had the opportunity to convey to the public the message that no complex technology could ever be accident-free, since equipment was bound to fail from time to time, and to explain the unprecedented efforts made to provide defence in depth against catastrophe.

5. A number of proposals had been heard for international action in other circles aimed at profiting from the Three Mile Island incident and reassuring public opinion, and these would receive the Agency's support. Three months after the accident there was increasing evidence that political and technical opinion was coming to appreciate the need for heightened international co-operation in nuclear safety research, the area in which NEA's own distinct contribution owed much to the Committee's collaboration and support. It was not that member countries' current efforts were insufficient; the essential need was to make it much more apparent that the efforts were in fact organised on a plurinational scale and that the results emerging were available to all those needing to know.

6. The Director General noted that changes of emphasis would certainly be required in the Agency's work in nuclear safety, and indicated that the Secretariat looked to the Committee for guidance on the necessary adjustments.

Briefing by the United States delegates on the Three Mile Island Accident

7. Dr. H.R. Denton, Director of the Office of Nuclear Reactor Regulation in the US Nuclear Regulatory Commission, and Dr. S. Levine, Vice-Chairman of the Committee, described the sequence of events during the accident. The following report incorporates particular pounts of interest and commentary that arose during discussion.

8. Dr. Denton said that at the time of the start of the incident, the reactor was operating at 97 per cent of full power. The initiating event was a loss of condensate flow brought about by a recurrent resin clogging problem in the condensate treatment unit, resulting in the main feedwater pumps tripping. (This trip had occurred two or three times a year in US reactor plants.) The auxiliary feedwater pumps started automatically and ran up to speed, but their isolating valves on the discharge side (steam generator inlet) were closed, and the steam generators rapidly dried out. It is instructive to compare, for different operating plants, the time taken for the steam generators to effectively dry out (no effective heat transfer) after all feedwater has been

lost. (The following dryout times for a given supplier indicate the ranges for configuration of plants now in operation. These dryout times are illustrative only, since differing assumptions were employed in their calculations.)

Shippingport	4 hours
Westinghouse plants	15-45 minutes
Combustion Engineering plants	10-15 minutes
Babcock & Wilcox plants like TMI (low inventory, once-through design)	
-Without anticipatory scram (pre-TMI)	less than 2 minutes
-Modified with anticipatory FW and/or turbine trip	5 minutes

9. The primary circuit pressure rose and the relief valve on the pressuriser opened at its set- 3-6 sec
point of 2255 psig (this valve had opened 152 times on US Babcock and Wilcox plants in 29 reactor-years of operation). As the pressure reached 2355 psig the reactor tripped on high pressure. Primary circuit pressure fell quickly (in about 10 seconds) to the closure setpoint for the pressuriser relief valve (2205 psig) but the valve remained open (this 13 sec
had happened three times before in B & W plants).

10. Temperature and steam pressure readings showed that the steam generators dried out in about 2 minutes. At the same time, the ECCS was activated automatically as primary circuit presure reached 124 sec
1600 psi. Unaware that the reactor was approaching a small LOCA as coolant vented through the stuck relief valve, the operator relied on the control room indication of pressuriser water level. (A safety valve associated with this relief valve had apparently been leaking coolant for some days, with the result that before the incident the indicated temperature in the discharge pipe of the relief valve was already high, as was the pressure in the quench tank. Also the position indication transmitted to the control room in fact reflected the

electrical signal being supplied to the valve, <u>not</u> its actual position.)

11. Since his instructions forbade him to allow the pressuriser to "go solid" (fill with water), the operator manually bypassed the ECCS signal when the apparent water level in the pressuriser reached the control setpoint. 3 min

12. The operator found the auxiliary FW pump valves shut and opened them. 8 min

13. As coolant continued to discharge through the relief valve into the reactor coolant drain tank (or "quench tank") the bursting disc on this tank blew at a pressure of 192 psig, allowing coolant to flow into the containment building sump. 15 min Since the containment isolation was related only to containment pressure (and not to ECCS actuation) the containment sump pumps transferred this split coolant to the radioactive waste holding tank in the auxiliar building. The sump pumps were turned off at 38 minutes. (Since the sump pumps were turned off before fuel became damaged, it is not believed that the reactor building sump was the source of high activity in the auxiliary building as previously stated. It is now believed that the high source of radioactivity in the auxiliary building resulted from leakage of the let-down system subsequent to core damage. Subsequently the operator switched off the primary coolant pumps since vibration and noise suggested they might suffer damage from cavitation. 100 min (Owing to the progressive loss of coolant through the stuck valve, a bubble of steam and gas above the core steadily expanded; the only (slight) cooling to the core was provided by running the primary coolant pumps. Once these were tripped, the presence of voids, unlnown to the operator, prevented natural circulation through the system, and core damage commenced.)

14. The operator identified the stuck relief valve and isolated it by means of the block valve. 2.3 hrs

15. Subsequently the operator actuated the ECCS intermittently, guided by the misleading indications of pressuriser water level. 3.5 hrs

16. The operator also re-opened the electromatic relief valve to depressurise the primary circuit to activate the RHR* at 400 psi. (The operator may not have been sufficiently aware of the potential effect of this depressurisation on the present voiding in the system. The depressurisation did, in fact, cause further voiding which resulted in an additional 7 hours of core being uncovered.) 7.5 hrs

17. A pressure spike to 28 psig occurred in the containment, thought to be due to hydrogen burning. 9h.50 min

18. The electromatic relief valve was closed to repressurise the primary coolant system. 13.5 hrs
Primary coolant pump A was started and cooling maintained through one steam generator to the main condenser. 16 hrs

19. Summing up, Dr. Denton said that during the time the core was uncovered, the temperature in the upper central region was at present thought to have exceeded 1800°C in the fuelled section, 1700°C in the unfuelled part. It seemed that early indications of high temperatures from some 50 thermocouples were disbelieved by the operator; the process computer read-out was not calibrated for temperatures above 700°F (371°C) and printed out question marks. In any event, the process computer printout was running 30 minutes late.

20. As mentioned earlier, the initiating event was loss of main feedwater, leading to the opening of the pressuriser relief valve, and non-availability of auxiliary feedwater should not itself have led to the accident. (In future, the setpoint for this valve to open will be raised, and the setpoint for the reactor trip on high pressure lowered.

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* residual heat removal system

Moreover, the reactor will automatically scram as a result of a loss of main feedwater and/or turbine trip.)

21. Tests had been done in this reactor design to demonstrate that cooling in natural circulation was feasible; however, during the accident the coolant was not subcooled and the operator was not initially aware of the presence of voids. (Currently NRC is evaluating the subcooling margin required prior to termination of ECCS.)

22. The implications of the siphon form of the surge line connecting the reactor to the pressuriser was also being studied.

Report by the United States delegates on the lessons learned so far from the Three Mile Island accident, and the remedial actions proposed as a result

23. Dr. Denton indicated that he would report under a number of headings:

(A) NRC orders to plant operators requiring immediate action;

(B) NRC orders relating to longer-term action;

(C) Short term recommendations (NRC staff positions) relating to plant operators;

(D) Short-term recommendations relating to design and analysis.

The following report includes points brought out in discussion.

24. (A) Orders requiring immediate action were issued in respect of nine plants (including Three Mile Island 1) built to Babcock and Wilcox design. These plants were shut down and the licensees required, as appropriate, to review and upgrade auxiliary feedwater reliability and performance, to implement operating procedures to initiate and control feedwater independently of the integrated control system, to wire a reactor trip ("control grade") dependent upon loss of main feedwater and/or turbine trip, to complete analysis of potential "small breaks" and formulate proper operating instructions, and to provide a senior reactor operator with TMI-2 training onthe B & W simulator, in the control room.

25. (B) Orders calling for longer-term action required the licensees to continue reviewing and upgrading the performance

of the auxiliary feedwater system, to submit a failure mode and effects analysis of the integrated control system to the NRC, to ultimately upgrade the reactor trip upon loss of main feedwater and/or a turbine trip to "safety grade", and to give continued attention to transient analysis and procedures for management of small breaks.

26. It was expected that the plants concerned, with the exception of those at Three Mile Island, would return to service during the summer.

27. (C) The short-term recommendations relating to plant operations were subdivided as follows:

Reactor operations management

28. It had been found at many plants that the shift supervisor had been overburdened with administrative tasks, spending as much as 70 per cent of his time on paperwork, staff problems, etc., to the extent that he might be unable to effectively oversee the continuing operation of the plant. A new command and control function was therefore being recommended, in order to free the shift supervisor of the administrative tasks and ensure that he was clearly recognised as being in command of all plant operations and decisions relating thereto.

29. It was also recommended that a shift technical advisor, employed by the utility, should be present at all times on each plant. The normal function of this specialist (who would presumably form a special relationship with the NRC resident inspector) would be to review the large number of licensee event reports, of which some 3000 were submitted to the NRC each year, and to propose safety measures at his home plant as appropriate. He would have no operations responsibility but would be totally familiar with the nature and status of the plant's safety systems and would advise the shift supervisor on all safety questions. In the event of an emergency the safety engineer would be expected to be capable of applying his technical understanding and his imagination to the particular situation so as to suggest suitable action rather than simply abiding by written instruction. It was possible that ultimately, as the

level of staff capability was improved, the job of the proposed safety engineer would be taken over by the senior reactor operator.

30. Dr. Denton acknowledged that this recommendation was perhaps specially suited to the US nuclear industry; with some 70 plants already in service, the operator capability was such that normal station shift staff would be unable to handle situations such as, for example, a small-break LOCA, lying between the design basis accident and a core melt, which were still not fully understood. These so-called "anomalous transients" were to be the subject of accelerated research and analysis (see paras. 52-56 below). It was acknowledged that the recommendation for a shift safety engineer reflected adversely on design concepts, but it has been concluded that it was justified on the basis of the TMI-2 accident having occurred after only 400 reactor-years of operation in the United States. Dr. Denton took the view that the US public was dissatisfied with the manning situation; it was hoped to attract high-grade engineers back to the utilities as soon as possible. On the other hand the appointment of a safety engineer should in future prevent utilities or others from losing sight of important incidents in other plants that could be generic in nature; he cited the 1977 incident at the Davis-Besse plant which began in the same way as that at Three Mile Island, differing in core burn-up (1 full-power day) and power level (9%).

31. It had been found that shift changeover procedures at Three Mile Island and many other plants had been slack, with marked lack of communication between succeeding shifts. One result had been that seven shifts had not remarked on the auxiliary feedwater trains being isolated. It was therefore recommended that an explicit procedure for shift changeover should be defined and implemented.

In-plant emergency preparations

32. Referring to the crowded control room at Three Mile Island during the incident, Dr. Denton pointed out that this was inevitable since the main control room had been the only

available assembly point for auxiliary system operators, maintenance staff and health physics teams. As a result it had been recommended to limit access to the control room during an emergency to the operations supervisory staff and to establish an on-site emergency operations centre containing a slave data printer and all necessary communications facilities. As-built structural drawings, piping and wiring diagrams, records, etc., would be assembled in the centre. An off-site emergency operations centre would be designated for each plant to provide for press and public briefings in the event of an emergency.

33. Consideration had been given to requiring constant voice recordings during any emergency for subsequent analysis, but the NRC Task Force concerned had not yet agreed on recommending it. Dr. Denton commented that merely recording all telephone conversations during the Three Mile Island incident had needed some 13000 tape cassettes which were now being transcribed.

Operational reliability and quality assurance

34. Dr. Denton indicated that it was proposed to make a radical change in licensing procedures in order to ensure that a utility was fully up to date with requirements for maintenance. It was being recommended that there should be a clear criterion for minimum plant quality (for example regarding the number of diesel generators or pumps available for service), with a mandatory plant shutdown if the criterion should be contravened, and a thorough enquiry by the utility management before restart. This new procedure would relate to major safety-related functions such as ECCS or electrical supplies; it was estimated that during 1978 the new procedure would have been invoked about 30 times had it been in force. As a further judtification, Dr. Denton reported that early in 1979 maintenance at the Three Mile Island plant had been reduced by one-third to cut costs. Only major safety functions would be covered, i.e. those which the plant licensee (technical specification) required to be operable while the plant was in service. The criterion was being set at loss of all redundant systems of one safety function. No shutdown would be required if half a safety function, including the

ECCS, were lost. This distinction was necessary since a number of trips were caused by instabilities arising from unreliable equipment, which would not themselves necessarily require the plant to be shut down. The new procedures would effectively place the burden of ensuring proper maintenance on the utility, which with shutdown costs running at two to five hundred thousand dollars a day might be expected to do more to prevent its plant becoming unsafe according to the new criteria.

35. Dr. Denton pointed out that the recommendations had been submitted to the Nuclear Regulatory Commission as a sufficient condition for allowing start-up of the temporarily shutdown Babcock and Wilcox plants; further recommendations to the Commission were in no way precluded. He announced that the Commission had now instructed him to implement these recommendations.

36. (D) The short-term recommendations relating to design and analysis covered all plants, and were subdivided as follows:

Provision of emergency power for "process" equipment

37. It was being recommended that emergency on-site power supplies should be provided for the pressuriser heaters, the pressuriser power-operated relief and block valves, and for the pressuriser level indicator. This is not a current requirement.

38. At Three Mile Island the pressuriser level indicator was not provided with an emergency supply since this parameter was not wired to trip the reactor. The pressuriser heaters were rated at about 2 MW, so the provision of an emergency supply might require installation of another diesel generator; however, the primary coolant pumps relied upon off-site power, and natural circulation could not be maintained if the heaters were not available. During the accident the heaters had tripped; not because off-site power was lost but owing to the fact that certain equipment and instruments had not been qualified to function in the wet containment environment and had been short-circuited by water.

39. Returning to the question of "anomalous transients" such as small-break LOCAs, Dr. Denton reported that some analysis had been done on maintaining core cooling without relying on the steam generators; this involved a feed and bleed approach and required operability of the pressuriser relief and block valves to be guaranteed. This requirement applied to all plants; as far as Babcock and Wilcox designed stations were concerned, about 15 per cent of the reported 152 lifts of the pressuriser relief valve (see para. 9) had been caused by deficiencies in the integrated pressure control system, and licensees of these plants were being required to install a mechanism separating the automatic system from the manual control.

<u>Performance testing of relief and safety valves</u>

40. The recommendation provided for these valves to be subjected to qualification tests under two-phase and water-solid flow conditions. In-service testing had not yet been decided upon but might be included as part of the implementation of the recommendation. Dr. Levine commented that the basis of the recommendation was the in-service history of these valves (see para. 9) which led to the probability of a small-break LOCA being an unacceptable 6×10^{-2} per year when account was taken of the other relevant probabilities in these plants. (Frequency of a transient with loss of feedwater: 3/year; pressuriser relief valve sticking open: 2×10^{-2}/year.) He pointed out that subsequent to the NRC bulletins requiring a reactor scram upon loss of main feedwater and/or a turbine trip, and raising of the pressure set point for opening the relief valve the probability of a small LOCA could be regarded as having declined to between 3×10^{-4} and 6×10^{-8}/year, according to the size of the relief valve and the capacity of the ECCS. (Here, frequency of a transient with loss of feedwater: 3/year; probability of restoring main feedwater within 10 min: 0.5-0.1; probability of auxiliary feedwater being unavailable when needed: 10^{-2} - 10^{-5}/year; probability of the relief valve sticking open: 2×10^{-2}/year.)

<u>Information to aid the operator during an accident</u>

41. These recommendations covered the provision of <u>direct</u>

indication of the true positions of relief and safety valves (rather than repeating the electrical signal being transmitted to them, as at present) and of instrumentation for detecting inadequate core cooling. The latter was still under intense investigation; it was not yet certain whether a wide range of in-core temperature indication would be called for. The primary need was a reliable method of water level indication, since it had been concluded that even with a properly designed surge line, a small-break LOCA could still occur with a spurious indication of a high level in the pressuriser.

42. In discussion Dr. Levine stated that consideration had been given to using computers for on-line diagnostics, which also covered disturbance analysis. The question of introducing noise analysis for diagnostic purposes was also being examined. As regards the man-process interface, he said that at present certain general criteria were applied to automatic systems; for example, if a system called for action to be taken in less than 10 minutes, it was automated, otherwise not. However, these criteria were liable to be modified as a result of the research programme on human error and performance. One aspect of the question of human interference with automatic systems had been dealt with in the NRC bulletins; typically the operator was prohibited from bypassing the ECCS until it had run for 20 minutes.

Containment isolation

43. Dr. Denton said that the Three Mile Island system for isolating the containment had in fact operated as designed, but the circumstances had shown the design to be inadequate and pointed the way to new needs. The system isolated only on a signal indicating high pressure in the containment; this had occurred at the 4 psig setpoint; when the pressure fell again the system was manually reset. If care had not been taken, this action could have enabled the sump pump to resume transfer of spilt water to the auxiliary building. A standard review plan in 1975 had modified this capability but the Three Mile Island plant, being already completed, had been excluded. The new recommendation was that the containment should isolate in

future on diverse actuation signals; for example high pressure, ECCS actuation or a third signal based on radiation measurements.

44. Although the only source of gaseous release was the liquids pumped to the auxiliary building, it had been found difficult to reduce the level of releases for about one week, owing to a number of leaks in the system.

45. It was also recommended that careful attention be paid to the design and qualification of the many essential and non-essential systems located in the containment. It was important for example to consider whether to isolate plain water systems when they contained highly radioactive water.

Post-accident containment hydrogen control

46. For those plants with installed hydrogen purge or recombiner capability, design review was being recommended to assure that reliable and particular containment penetrations have been provided for hydrogen purge and recombiner systems, and that they would not be vulnerable to single failures. In view of the potential for hydrogen generation, the requirement for all Mk I and II containments to be inerted was being renewed, and it was recommended (minority view) that all plants should have the capability for installing and using recombiners at one or two days notice.

47. In discussion Dr. Denton said that although there had been some support for installing a venting system on the primary circuit the NRC Task Force involved had not recommended it; similarly the proposal was for a recombiner capability rather than for recombiners to be installed, since they were slow in operation. The recommendations were preliminary and it was possible that they would be expanded to provide, for example, for additional monitoring of the containment atmosphere or for a fission gas absorption capability in the recombiner line. Purging was not a viable proposition in the United States except as a last resort, owing to considerable public concern about releases of any kind. During the process of venting the primary circuit to the containment all available fans and blowers were operated, but it was clear that some hydrogen pocketing did occur because of the burning that took place; no

detailed calculations had yet been done to study its possible extent.

Radiation control for systems outside the containment

48. This recommendation was for leak tests and a shielding review to be done for safety and process systems located outside the containment. In particular the residual heat removal system was now considered to be part of a line of defence (third barrier); it had not been previously regarded as such. It also merited special attention because it now probably contained highly contaminated water.

Auxiliary feedwater system reliability

49. NRC orders had already required licensees to upgrade auxiliary feedwater system reliability (see paras. 24-25); this recommendation covered requirements for automatic actuation of the system and for direct flow indication.

50. Dr. Levine commented that if relief valves were too small to permit adequate feed and bleed flow, the core could not be properly cooled using the ECCS alone in the event of a small LOCA. It would be necessary to use the auxiliary feedwater system to help remove heat, and hence its reliability must be high. Remarking that a plant with only one steam-driven auxiliary feedwater pump per unit (such as Oconee 1) was a very low reliability system, Dr. Denton reported that NRC was requiring unit auxiliary feedwater systems to be coupled and a station operator to be continually present. Dual electrically-driven auxiliary feedwater pumps were being added to the plants concerned and would be in service by about the end of August 1979. Consideration would be given to the potential for rapidly cooling the primary system using the steam generators and the auxiliary feedwater system.

Instruments to follow an accident

51. The recommendations were for improved instrumentation for post-accident sampling, effluent monitoring, containment monitoring (high range) and for in-plant and effluent iodine measurement. They represented improvements to an existing NRC guide (No. 1.97). At Three Mile Island the utility was already

erecting a separate building for processing contaminated water; it was possible that some changes would be recommended for effluent treatment systems at other plants.

Analysis and design and off-design events for improved training and procedures

52. Operators had hitherto been trained on simulators using procedures written for the design basis accident (DBA). It was now believed that operator training should go further and cover small-break LOCAs, core uncovering such as occurred at Three Mile Island, and the so-called multiple failure events. (It was noted that these proposals referred only to training, not to the design basis.) The NRC was satisfied that in Babcock & Wilcox plants two high pressure injection systems would always be sufficient to cool the core without assistance from the steam generators. However, it had become clear that past analysis of small LOCAs had been inadequate for situations where pressurisation was in competition with other factors. In conclusion, Dr. Denton said that the changes being recommended were selective and fell short of a requirement that all plants should be shut down; there was no feeling that reactors were unsafe and deserved to be shut down.

53. Dr. Levine described the impact of the accident on nuclear safety research in the United States. Generally speaking it could be said that the outlook was certainly not "business as usual"; the NRC had been requested by the Advisory Committee on Reactor Safeguards to take certain specific actions by given deadlines. It now seemed that the "envelope" approach to safety used hitherto needed to be re-examined.

54. The principal technical lesson learned from the Three Mile Island accident was that accident situations involving severely damaged cores which lie between the design basis and a core melt needed careful study. In the FY-79 budget some $12 million was being spent on TMI-related research; a request had been made for approximately an additional $30 million for the 1980 budget.

55. Detailed codes were needed to predict these intermediate phenomena as precisely as possible, but in parallel, faster and

simpler codes were needed to provide rapid understanding of plant behaviour in these so far less explored areas. Codes such as TRAC, developed for large LOCAs, were now being modified to handle variations in secondary system performance. All these new and modified codes would require experimental support from facilities such as LOFT and Semiscale for the development of physical models. Discussions were already in hand with a number of other countries regarding possible complementary studies to supply further data.

56. These codes would also serve to assist studies aimed at training operators to deal competently with accidents beyond the design basis; it was necessary to investigate the instrumentation needed by operators to enable them to understand and react properly to a wide range of postulated accidents. This would cover display and diagnostic systems such as had been developed at the OECD Halden Reactor Project. The codes would also be used for analysing transients which had already occurred in operating reactors, and thus provide the information needed for conducting further transients in operating reactors.

57. It was proposed to do further work on risk assessment to construct event trees for accidents in the area lying between the design basis accident and a core melt, as a guide for code development. Similarly, much more effort would be devoted to the potential effects of human error on the source of this kind of accident. These two tasks were obviously closely related.

58. A small programme already in hand outside the area of confirmatory research was directed towards finding ways of improving reactor safety; additional funds would be sought to extend this programme specifically as regards PAHR, vented containment and emergency core cooling.

59. Other areas where improved understanding was essential were primary coolant chemistry following severe fuel damage; the evolution and behaviour of hydrogen in the primary coolant circuit and containment; long-term behaviour of important plant components under accident conditions, for example the primary coolant pumps were not required to operate in a LOCA environment yet were important in preventing an accident from deteriorating.

60. Finally the NRC has recognised the importance of preserving the Three Mile Island data; it was necessary to investigate the radioactive materials in the containment, to assess the nature and extent of core damage and also the state of the safety equipment. Discussions were already going on with the Department of Energy and EPRI about criteria for re-evaluating the safety of the plant. Dr. Levine indicated that all this information would be made available to member countries in the usual way.

61. The following points emerged during discussion:

(i) The work proposed on primary coolant chemistry and hydrogen would be mainly theoretical, since a great deal of data was already available. However, more experimental work was not precluded, should more information prove necessary.

(ii) The established test series on large LOCAs in the LOFT facility would probably be interrupted soon for a number of small LOCA tests to be conducted.

(iii) The system scramming the reactor upon a turbine trip was safety-related and redundant (before the accident most US plants except those designed by Babcock and Wilcox had this sytem; the requirement might be extended to include other plants).

(iv) The codes to be used for small LOCA studies differed from those already used in a number of ways: variations in plant parameters (presence of a secondary system to control steam generator conditions, operator interference, etc.), additional physical parameters and presence of non-condensible gases.

(v) The burden of work related to the accident was such that there would probably be some changes made to the licensing organisation. Assignment of staff within NRC to the different task forces was naturally making it difficult for staff to keep up with daily licensing tasks.

(vi) The aftermath of the accident would probably cause some change in the accident fuel studies in the PBF facility, for example if more data on metal/water reactions proved necessary, but the programme changes, as yet undefined, would be small.

(vii) There was agreement in principle about a programme to look at core damage at Three Mile Island. Planning had just begun but no details of ways and means, timing or location had yet been defined.

(viii) Returning to the question of collecting information about operating experience, Dr. Levine said that it was being recommended that an operations evaluation function should be established in order to evaluate reactor operations experience. During the last two years the NRC had analysed the information considered as having safety significance, essentially the data on failure rates. It seemed certain that the data which had not been considered as having safety significance did in fact contain some precursors of the Three Mile Island accident that would have pointed valuable lessons if properly studied. Mr. Stadie (Secretary) referred to the Committee's decision at its meeting in November 1978 to set up a Working Group to prepare an outline for a centralised system for reporting and analysing information about incidents and malfunctions in nuclear plants, on the basis of a United States proposal. The Working Group had in fact held its first meeting on 27th and 28th March; since most of the people concerned had become heavily involved in the aftermath of the Three Mile Island incident, it had not yet been possible to make much progress. Nevertheless, it was hoped to put a proposal to the Committee at its next meeting.

(ix) Following a question about control rod material having the lowest melting point of any materials present in the core, Dr. Levine said this would be examined.

(x) Referring to the technical support group which had been assembled for Three Mile Island, Dr. Levine concluded that the principal problem had been that of obtaining information about plant status, systems behaviour and so forth, so coherent analysis had been very difficult. This had been one of the main reasons for advocating improved data display and diagnostic capability to assist the operator.

Reports from delegates regarding present thoughts about the influence of the Three Mile Island incident on national programmes and practices

62. The Chairman invited delegates in turn to give their views and factual reports. (There was naturally some common ground in the reports on changes made or foreseen in national programmes, therefore this record, which reflects only the statements made, summarises the proceedings under a number of appropriate headings. Delegates' replies to questions, and salient points of the discussion, are incorporated.)

63. All countries having active nuclear power programmes involving light water reactors (Belgium, Finland, France, FR of Germany, Italy, Japan, Netherlands, Spain, Sweden, Switzerland) took immediate steps to set up expert bodies with the task of reviewing the safety of their plants, particularly those already in service, in the light of the information provided by the USNRC on the features of the Three Mile Island (TMI) accident. Most of these countries also sent technical missions to the TMI plant to study the question at first hand in consultation with NRC representatives. The questions to which urgent attention was being given by most countries were naturally those reported by Dr. Denton as receiving most effort in the United States, or others connected with or derived from them. (See paras. 8-60 above.)

64. A number of delegates indicated that review of the Three Mile Island accident and details of the plant design had shown that some of the actions taken by the NRC in response to the accident were not applicable to their reactors. However, all delegates from countries having active nuclear power programmes utilising light water reactors indicated that one or more of these measures, and certain others, were planned or had already been taken in respect of the reactors in their

countries. These measures and safety considerations may be sub-divided into three areas: plant design modifications; procedural and administrative controls; safety analysis and evaluation.

Plant design modifications

65. All countries having light water PWRs had required, or were reviewing the need to require, a change in ECCS activation from a low pressure/low level coincidence signal to a signal based upon low pressure only since a small LOCA could result in a pressure decrease not accompanied by a significant decrease in pressuriser level. In several countries it was intended to modify the signal initiating containment isolation such that isolation would occur if the ECCS should be activated. A number of countries were also considering adding an additional containment isolation signal based upon a high radiation level in the containment. In one country the automatic start feature of the containment sump pump had been removed.

66. Two countries had already required plant modifications to provide a capability for installing a hydrogen recombiner. However the question was raised that with the apparently very high core temperatures experienced during the TMI accident, the hydrogen production rate could be far in excess of the removal capability of a recombiner.

67. Several countries were reviewing the storage capabilities for liquid and gaseous effluents that might be highly radioactive, and the ability of the radwaste system to process them. It was indicated that changes would be made as deemed necessary.

68. The Committee was informed of certain other plant modifications which had been made or were under active consideration: on one plant, the block valve upstream of the pressuriser power-operated relief valve (PORV) had been provided with an automatic closure signal based on low pressuriser pressure so as to prevent the primary circuit from depressurising should the PORV fail to reclose; in another plant the set-point for the high pressure reactor trip had been lowered to prevent the PROV from lifting at all during a transient; several countries

were reviewing the need for improving the indication of valve positions and control room alarms for valves out of their required positions; for better systems monitoring reactor coolant leakages, for primary circuit venting systems to prevent gas buildup, and a network of more effective in-plant and near-plant radiation monitors.

Procedural and administrative controls

69. As noted above, delegates from all countries with active LWR programmes had set up special commissions or had undertaken special reviews to incorporate the lessons learned in the Three Mile Island accident. A step taken by nearly all countries was to carry out a detailed special inspection of the test and maintenance procedures in use at their plants to ensure that safety equipment is properly restored to ready status following test or maintenance.

70. In several countries, nuclear power plant operators had been given additional training in dealing with accidents of the TMI type to ensure that they had absorbed the lessons of that accident. Post-accident procedures had been thoroughly reviewed in most countries to ensure that instructions for diagnosing plant abnormalities were adequate and that the conditions in which ECCS equipment might be stopped, after starting automatically, were clearly understood.

71. Certain countries had already undertaken a review of their emergency schemes to ensure that events of the TMI type were covered.

72. A number of delegates indicated that procedures for shift changing in their nuclear power plants were being reviewed to ensure that all appropriate information regarding plant conditions were passed on. In particular, one country had already introduced a special checklist to be used before a reactor startup to verify that all valves important to reactor safety were in fact in the desired position.

Safety analysis and evaluation

73. In addition to the above immediate actions, several longer range questions were reported to be under discussion.

74. Nearly all delegates foresaw the TMI accident leading to some changes in their nuclear safety research programmes. In one country for example, licensees had been requested to perform additional LOCA analyses incorporating the presence of gas or steam pockets, and to analyse the behaviour of reactor coolant pumps in two-phase operation. Several delegates expressed the view that the accident had pointed out the need for analysis of the effects of a total loss of safety function, such as a loss of all plant electrical power or a loss of the ultimate heat sink. The purpose of such studies would be to determine the approximate time for the start of significant core damage.

75. A number of delegates indicated that the accident pointed out the desirability for a more accurate, or more direct method of determining primary circuit inventory, perhaps by direct measurement of the water level in the reactor pressure vessel. Several countries reported that they were evaluating such methods.

76. One country was evaluating the need for a fast-acting system for detecting fuel clad rupture, in the light of experience at TMI.

77. The following additional topics were being evaluated: the need for better control room instrumentation to display parameters of particular interest during an accident; the potential radiation levels to be expected in the vicinity of safety-related equipment following an accident; improvements that might be needed to auxiliary buildings to prevent release of radioactivity; changes to the qualification requirements for equipment both inside and outside the containment to improve their ability to withstand an accident; and changes to relief valve qualification and testing to improve reliability.

Other reactions to the accident

78. The delegate from the United Kingdom indicated that although PWR plants were not in use, a number of general lessons had been drawn which would affect licensees in the longer term; These included reviews of emergency schemes, operating practice

including training and communications in the event of an accident. It was noted than an engineer responsible only for plant safety and independent from the plant operating staff had always been assigned to each operating nuclear plant.

79. A number of delegates, notably from smaller countries, referred to the financial and administrative problem for countries having less well developed infrastructure in coping with an accident. Authorities were studying this problem with some urgency.

80. The representative of the Commission of the European Communities reported on the actions planned by the Commission in response to the Three Mile Island accident. These included: reinforcement of emergency preparedness at the Community level; the establishment of a group of high-level experts to examine safety regulations and standards in member states with a view to harmonising them; and the feasibility of a Community system for exchanging information on accidents. In addition, the pluri-annual programme of the Ispra Joint Research Centre was under review, and additional effort on blow-down experiments was likely.

81. Most countries reported a hardening of the public attitude to nuclear power following the accident, with opponent groups marshalling sophisticated arguments. However, all observed the fickleness of public opinion when new events drove the accident off the front page, and the general view was that for countries considering introducing nuclear power the result of the accident could be some degree of delay.

Discussion of the possible impact of the Three Mile Island accident on the CSNI's programme of work

82. Referring to the reports by delegates, the Chairman stated that although it was too early to draw detailed conclusions, signs were already emerging of some consensus about how safety concepts and review procedures might change. Some initial conclusions might be possible at the November meeting of the Committee.

83. Following a proposal by the delegate from the FR

Germany, the Committee agreed that a Working Group, consisting of representatives from those countries preparing national reports on the TMI-2 implications, should be set up to prepare a consolidated report for consideration by the CSNI at its November meeting. This report would provide a factual review of the some 20 major issues stemming from the TMI-2 accident which have relevance to the CSNI activities.

Current views about quantitative risk assessment in the light of the Three Mile Island incident

84. The Chairman (Prof. A . Birkofer, FRG) spoke of the general attitude to quantitative risk assessment (QRA) and gave his view that its deficiencies had certainly been overstressed. Following the Three Mile Island accident, critics in the FR of Germany had become very active and had attacked QRA as a general approach; in fact, QRA techniques, especially event and fault trees, provided an important method of identifying potential weak points in safety-related circuits and systems, in particular for evaluating the interface between different engineering disciplines. He indicated that some pre-publication results of the German reactor safety study had resulted in certain modifications being made in his country's PWR power plants. He invited Mr. F.R. Farmer (UK), former Chairman, to give his views about the current status of QRA.

85. Mr. Farmer said that a number of issues had confused the QRA picture in the past. A major objection to its use had been the view that its answers were perhaps better not known; decisions taken on the basis of QRA would become publicly known, with the result that the calculated risks (of equipment failure for example) would either be challenged as inaccurate or assumed to have been accepted. He took the view that in time progress in QRA would be on the basis of what levels of risk were unacceptable, rather than on that of some risks being acceptable. He cited the British risk study of the petrochemical complex on Canvey Island, in which certain risk high spots (as assessed) had been judged to be unacceptable, after which ways and means of reducing them by a factor of two or three had been devised.

86. In the nuclear industry, opponents of QRA had long said that it should not be attempted, either because it would lead

ultimately to the inescapable and difficult final decision, or because it was not precise. The nuclear community acknowledged the lack of precision, but believed that QRA enabled otherwise impossible comparative judgments to be made between technical options, and invariably improved understanding of the system being considered.

87. With regard to the selection of areas for applying QRA, Mr. Farmer thought that much more should be done on complex interrelated systems whose failure rates were predictable within a factor of two or three. United Kingdom studies were making substantial headway in the nuclear and chemical industries (he pointed out that the latter had potential for catastrophic accidents leading to a number of deaths in the range 1.000 to 10.000). He believed that during the next five years the safety reports which industrial firms were required to produce would develop in a quantitative style - again, imprecise, but permitting comparative judgments to be made.

88. In conclusion, Mr. Farmer thought that although QRA was making progress, there were a number of pitfalls in the way. However, it was being increasingly accepted by technical organisations as a means of clarifying issues where different options were present, even when uncertainty persisted about the precision of the results.

89. Dr. Levine reported that the use of QRA techniques by the USNRC would be increased as fast as possible. He pointed out that the Reactor Safety Study had not in fact been repudiated by the Nuclear Regulatory Commission, who had endorsed the approach that procedures should be developed to apply QRA in the regulatory process. A draft paper to this end was being prepared. A number of broad-based studies, re-examining the technical framework of the licensing process, were in hand, and he considered that the accident had merely emphasised the need for greater use of QRA methods.

90. In discussion about other possible approaches, Dr. Levine took the view that the well-known "envelope" approach involving the design basis accident (DBA) had served well and had had a good safety record despite the Three Mile Island

accident. However, the outlying events which did not fit the general pattern were a considerable problem. By applying event tree/fault tree analysis to a number of different plants (not all), the inter-relationships between systems could be redefined, and a more sophisticated alternative to the single failure criterion found. This technique, based upon QRA, could then be applied universally, with no need for studies of each individual plant.

91. Mr. Farmer commented that future studies of this kind would not necessarily be as comprehensive as the US Reactor Safety Study. The amount of time spent on each study was now the most significant question, and plant studies could therefore be broken down into examinations of basic sub-systems such as ECCS or emergency electrical supplies. His view was that in the long run engineers would be required to know enough about reliability engineering to be able to design systems with a low probability of failure. This trend was already visible in the British chemical industry.

92. Mr. Togo (Japan) said he would like to propose two subjects related to QRA for further study in the light of the accident; first a clarification of the degree of conservatism in current safety evaluations involving the design basis accident, sub-divided into the accident postulation, the accident sequence, and (his principal concern) the dispersion of fission products. Secondly, he thought an effort should be made to evaluate the extent to which plant reliability could be enhanced by the different proposed improvements in administrative controls. Following a suggestion from the Chairman, Mr. Togo undertook to send his proposals in writing to the Secretariat in good time before the meeting of the Bureau to be held in September.

Closure of the Meeting

93. The Chairman reiterated the Committee's gratitude to Dr. Denton and Dr. Levine for their comprehensive briefing and for their frank and helpful answers to delegates' questions. There being no other business he closed the meeting, confirming that the regular meeting of the Committee would take place in Paris on the 13th and 14th November 1979, with the sub-Committee

on Licensing meeting on the next day as usual.

SPECIAL MEETING OF THE SUB-COMMITTEE ON LICENSING 29 JUNE 1979

94. A special meeting of the sub-Committee on Licensing was held the day after the main CSNI meeting to discuss implications of the TMI-2 accident on licensing and regulatory matters.

IAEA's activities

95. Mr. Rosen (IAEA) presented a brief report on the Agency's plan in the light of TMI. He referred to actions taken by IAEA immediately after the incident, first to collect information at TMI and then to solicit expert advice as to the impact of TMI on the Agency's programme. Subsequently an increased programme has been proposed to the IAEA's Board of Governors, covering additional activities in the following fields:

(a) In the NUSS programme, preparation of additional safety guides are being considered.

(b) An international nuclear safety meeting of a general nature is planned for mid-1980.

(c) Assistance measures are being considered in case of accidents in countries which may have difficulties in handling them in view of their limited capabilities, taking account of legal, liability and financial aspects.

(d) General advisory services in the form of missions and assistance for review are planned in order to increase national regulatory capabilities when needed.

Altogether this would result in an increase of 40-60% of the Agency's programme on nuclear safety.

96. An extensive discussion followed, raising doubts about the efficiency of a prompt action from an assistance team composed of foreign experts to help in "stabilising" the reactor during the initial phase of an accident. The general opinion

was that for the first period of an accident (half a day to 1 day) sufficient expertise should be available at the site or nearby and has therefore to be within the resources of the country where the accident happens. In this respect it was suggested that data links between the plant control room and outside teams might be more efficient than sending additional experts to the site.

97. For longer term, after stabilisation of the reactor there is an obvious need for assistance from outside, particularly in the case of radioactivity releases where sampling and analysis equipment is required.

98. It was concluded that a distinction should be made between action to be taken on the site immediately after an accident and the "recovery" measures to be taken outside the plant which have to rely on the existence of emergency plans and external assistance.

Present thoughts about the influence of the TMI incident on nationa licensing practices

99. Several delegates expressed preliminary views about the impact of the TMI accident on nuclear safety in general, or more specifically on their respective national programmes and licensing systems, pointing out the need and the difficulty in keeping nuclear risks in perspective. In several countries, action is being taken to re-examine the nuclear safety philosophy and procedures in the light of TMI and to look for improved methods of informing the public and gaining credibility.

100. In particular, in France technical aspects such as a more systematic study of initiating events, the concept of the Design Basis Accident and reliability of systems are being reviewed to identify possible gaps and improvements. In the FR of Germany, in addition to a reanalysis of the safety reports on power plants either in operation, under partial licence or under construction, a review of the nuclear safety philosophy is planned in order to see notably whether developments are needed to limit the impact of low probability-high consequence accidents not covered by the present Design Basis Accident concept, and to discuss such issues as for example underground siting.

101. However, no drastic changes are expected in spite of likely developments both in technical aspects and in administrative procedures such as in licensing and emergency planning. Some delegates expressed their concern about a possible tendency in the future to ask for safety measures which might not take properly into account cost/benefit considerations and therefore may appear as "unreasonable" requirements.

102. Following this discussion, it was agreed that on the proposal of the Delegate from Finland, it would be useful for a brief synopsis (about two pages) of each country's position after the TMI accident to be addressed to the Secretariat within approximately two weeks and then distributed to the Sub-Committee. (Finland, Italy and the USA have already circulated written statements during the CSNI meeting and are therefore not requested to submit additional information.)

Possible impact of the TMI incident on the Sub-Committee's programme of work

103. At this preliminary stage it was not posible to discuss this topic in detail and the Chairman suggested that the possible implications of TMI on the Sub-Committee's programme might be discussed during the enlarged Bureau meeting of the CSNI in September and reported at the next meeting in November 1979 for discussion. This suggested was adopted.

Other business

(a) Preparation of the Specialist Meeting on the Regulatory Review in the Licensing Process (Madrid, 7-9th November 1979)

104. Mr. Giuliani, Chairman of the Working Group on Regulatory Review, reported the progress made in the preparation of the specialist meeting, indicating that about 20 abstracts had been received so far for consideration by the Group. He suggested that a session be devoted to the impact of TMI on regulatory review and that countries could be given the opportunity to present their respective situation at the meeting. The Sub-Committee agreed with this suggestion and instructed the Working Group to make arrangements for such a session during the specialist meeting.

(b) Emergency planning activities within the NEA Programme

105. The Chairman announced that the Committee on Radiation Protection and Public Health (CRPPH) had put on the agenda of its next meeting in September, an item on emergency planning on the occasion of the discussion on the TMI situation. A discussion followed concerning the possible role of the Sub-Committee in this field and the interface between emergency action at the plant site and the public health and administrative measures to reduce radition dose to the population and environment around the site. In this respect, it was noted that in the TMI case consideration for evacuating local population was based essentially on the potential within the plant for a hydrogen explosion which may have resulted in a significant release of radioactivity and not on radiation exposure levels. The possibility of setting up a joint group with CRPPH was mentioned to look into present practices and emergency requirements in case of accidents.

106. Finally it was decided that an observer for the Sub-Committee should attend the discussion of the CRPPH meeting on this item, and that the subject should be discussed further at the enlarged Bureau meeting of the CSNI and reported at the next regular meeting of the Sub-Committee in November 1979.

PART II

SUPPORTING VISITS

Discussions with Dr. M. Tanguy, Director of the "Institute de Protection et de Surete Nucleaire" - The Institute of Protection and Nuclear Safety (IPSN) of the CEA - Fontenay-aux-Roses, 26 June

107. The main purpose of this meeting was to:

(i) explore in depth the licensing and regulatory procedures in France for the control of nuclear installations and in particular the research reactors and front end activities of the nuclear fuel cycle; and

(ii) the possibility of establishing closer links between the AAEC and IPSN on nuclear safety matters.

108. The following officers in the Institute were involved in the discussions:

Dr. M. Tanguy, Director of IPSN
M. Monteil, in the Administration Branch
M. Jacqmar, in the Department of Nuclear Safety.

The Australian delegation consisted of:

Mr. D. Crancher, Director of the Nuclear Plant Safety Unit, AAEC, and
Mr. C.R. Jones, Minister, Australian Embassy, Paris.

109. The IPSN officers explained that statutory intervention by Government authorities in France in regard to nuclear safety is exercised in three principal complementary ways:

. a system of individual licences for each installation after a thorough technical examination of the proposed safety measures;

. the drawing-up and application of general technical rules and regulations concerning safety;

. inspection.

Ministry of Industry

110. In France, the Ministry of Industry is responsible for matters concerning the safety of nuclear plants, and is the licensing authority per se. It discharges this responsibility through the Central Safety Department for Nuclear Installations (Service Central de Surete des Installations Nucleaires - SCSIN).

The Institute of Protection and Nuclear Safety (IPSN)

111. This is a part of the CEA (the Atomic Energy Commission). It employs some 850 persons deployed in two major departments - the Department of Nuclear Safety (Department de Surete Nucleaire (DSN)) and the Department of Protection (Department de Protection (DP)).

112. All licence applications are made in the first instance to the Ministry of Industry for processing by the SCSIN. The SCSIN has only a relatively small technical competence and all licence applications are reviewed in depth for the Ministry by IPSN. The IPSN is in effect the technical wing of the Ministry. In its decisions the SCSIN relies heavily upon the IPSN report.

113. Licensing procedures apply to:

(i) nuclear reactors, including those used for research and isotope production;

(ii) particle accelerators capable of transmitting more than 300 MeV energy;

(iii) plants for the preparation, manufacture or conversion of radioactive substances, particularly plants for the preparation of irradiated nuclear fuels or the processing of radioactive waste; and

(iv) facilities for long term storing or the use of radioactive materials, including radioactive waste.

114. Dr. Tanguy stressed the independence of IPSN within the CEA. The establishment of IPSN within the CEA was administratively very convenient, since it gave the IPSN the full technical support of the expertise within the CEA. Staff within IPSN were drawn from CEA establishments and were mobile. However, in regard to advice tendered on licensing applications, the Director of IPSN was not accountable to the CEA but to the Department of Industry, thus ensuring complete independence within the licensing arrangements. This was considered an important issue, particularly in regard to the licensing of the CEA's own plant.

115. There was a detailed discussion on possible means of improving contact between the IPSN and the AAEC. The discussion focused on both the broader issue of exchanges between the two Commissions on nuclear safety matters in general, and the needs of the AAEC's new reactor study.

116. It was agreed by both sides that contact should be

restricted to the IPSN/L & R Bureau. In fact Dr. Tanguy stressed his reluctance to establish direct links with project groups within the AAEC, such as the new reactor study group; he preferred to restrict dialogue to the L & R Bureau.

117. It was finally agreed that regular contacts in the field of nuclear plant safety, public health and environmental protection should be established between the L & R Bureau and IPSN. Dr. Tanguy agreed that at this stage there would be no need to formalise this arrangement outside of the 1969 general agreement on Scientific and Technical Co-operation between the AAEC and CEA. However, he believed that the arrangement should be formalised, under the terms of the 1969 agreement, by an exchange of letters between the AAEC and CEA senior managements.

118. Within the Australian Atomic Energy Commission the responsibility for these areas is with the Licensing and Regulatory Bureau within the Regulatory and External Relations Brancy. The structure and responsibilities of the Licensing and Regulatory Bureau is described in the enclosed documents.

119. All correspondence and exchanges on scientific and technical matters under this arrangement would accordingly be between:

Mr. D.W. Crancher, Director of Nuclear Plant Safety (corresponding to Department De Surete Nucleaire); and
Mr. J.E. Cook, Director of Environment and Public Health (corresponding to Department de Protection).

120. It is understood that these contacts would be confined to exchanges of a special nature under the terms of the 1969 agreement. They are not supplementary or prejudicial to any existing contractual agreement between the AAEC and CEA relating to specific projects. Specific arrangements relating to particular projects would be matters for further discussion between the AAEC and the IPSN.

121. I undertook to pursue this question in Sydney. Mr. Jones, Minister, Australian Embassy, was extremely helpful in providing guidance on essential policy matters during the discussion.

Discussions in the UK with Mr. O'Neil of the of the Safety and Reliability Directorate (SRD), UKAEA, 2 July 1979

122. Discussions were held with Mr. R. O'Neil of the SRD on 2 July 1979. The following topics were covered.

Safety assessment techniques for reactors, including nuclear propelled ships

123. I queried with Mr. O'Neil the results of an assessment carried out by his staff of Australian ports for nuclear powered ships. In particular, the case of an arbitrary cut-off at 10 rem in the estimation of population thyroid dose. This has led to conclusions regarding the acceptability of some large ports, such as Sydney, quite contrary to those of the AAEC which used a much lower cut-off.

124. O'Neil agreed that the 10 rem shut-off was not defensible and that current SRD assessments of population exposure were extended to a cut-off of 1/2 background.

125. The SRD Code TIRION, which had been developed for atmospheric dispersion calculations, had a very wide acceptance both in the UK and overseas. It was even used by many nuclear opposition groups such as the FOE. O'Neil recommended that the AAEC should consider adopting this Code.

126. SRD was now closely assessing possible consequences of Cs 137 fallout from releases. This may have been given insufficient attention in earlier studies.

Noise detection

127. Developments in noise detection techniques for pressure plant integrity were also discussed. He considers that periodic in-service inspection may eventually be replaced by permanently installed noise detection instrumentation.

Discussion with Officers of the Nuclear Installation Inspectorate (NII), July 4, 1979

128. Discussions were held with Mr. Gausden, Chief Inspector, and Messrs. McLoud, Wood and Charlesworth.

IMCO Code

129. The NII is now involved with reviewing the IMCO Code

on nuclear ship safety, and it is apparent they are far from pleased with it. The extent of the jurisdiction of the NII on nuclear ships is not entirely clear at this stage. However, they are currently assuming responsibility for reactor safety under the Health and Safety at Work Act, which gives them wide powers for protecting both workers and the general public.

130. Mr. Wood maintained that the NII is concerned with compatibility of the IMCO reactor safety standards with their own civil reactor standards. The low level of ship safety (i.e. collision protection, grounding, etc.) was also a matter of concern. In general, nuclear ships coming into UK ports under continuous trading conditions would be expected to comply both in regard to berth siting and nuclear plant safety with their civil reactor requirements for siting and safety.

Reactor Safety Principles

131. Wood confirmed that the NII safety principles were applicable to reactor systems other than the gas-cooled power reactors, although they had evolved around these. The Fast Reactor would in fact be reviewed against these principles.

132. I explained that we were attempting to adapt them to research reactors and he confirmed that this would be appropriate. In carrying out assessments of nuclear plant, the NII first review the engineering quality, i.e. choice and compliance with relevant standards, good practice, design procedures, etc. This was followed by a system analysis, starting with single failures (fires, LOCA, both small and large, etc.), working up to an analysis of more complex systems using event trees backed by fault trees where necessary.

133. In assessing system failures, a combined failure of process equipment plus protection less than 10^{-7} failure on demand would be considered incredible. This was the CEGB requirement.

134. The ERL currently used by the NII for reactor safety assessment were based on twice the DEL for occupational exposure, e.g. 10 rems W.B. and 30 rems thyroid.

135. The concept of "barrier" used in the NII safety principles went beyond the single concept of the physical barrier, used in US safety arguments. Essentially the NII included all those systems and protection requirements required to maintain physical barriers.

Research Reactors

136. The NII jurisdiction over research reactors was covered by two Acts. Privately-owned research reactors, such as those operated by the universities, were licensed and regulated by the NII under the Nuclear Installations Act of 1965, as amended.

137. The UKAEA reactors (DIDO, PLUTO, etc.) were subject to the control of the NII by regulation (not licence) under the Health and Safety at Work Act. Under this Act it was now a duty to ensure the safety of people (i.e. the public) and not just employees as under the old Factory Act. Under this Act the NII can exercise controls by regulations under the Act backed by guidelines and codes of practice. The NII has had very little involvement with UKAEA reactors, mainly because of lack of staff but also because of the good safety record of these plants which has not justified an urgent involvement. However, this situation is ever-changing, and the Northern Group of the NII is now commencing a review of these plants. (N.B. The AAEC might need to consider the advisability of opening a dialogue with the NII on this matter.)

Training of Staff

138. Mr. Gausden, the Chief Inspector, repeated his offer, made some time ago, to accept a Commission engineer for a two year attachment. The engineer would be fully integrated with the NII but the NII would be sympathetic to AAEC preferences where the officer should be attached.

Discussions with the Canadian Atomic Energy Control Board (AECB) Ottawa, 6 July 1979

139. Discussions were held with the following officers;

Mr. R.W. Blackburn, Secretary, AECB
Mr. Z. Donnaratzki, Director, Reactor & Accelerator Branch
Mr. A.B. Dovey, Manager, Uranium Mine Division.

Three Mile Island Accident

140. The Canadian response to TMI-2 appeared to follow world trend with excessive media headlining and intense public awareness at all levels. In general, editorial responses were far more restrained and responsible than headline reaction. Canadians were sensitive to trans-boundary fallout risks.

141. One consequence of TMI-2 has been a proliferation of public enquiries at both State and National level. The State of Ontario had now instituted a public enquiry into nuclear power and the AECB was submitting evidence on CANDU safety. Critics of nuclear power were submitting evidence critiscising the CANDU ECCS provisions (I recalled the AAEC's own adverse comments on this at the time of Jervis Bay). Since 30% of the total generating capacity of Ontario Hydro was now nuclear, the outcome of the enquiry was of some concern.

142. The AECB was somewhat critical of the close involvement of the NRC with the TMI-2 incident. They believe its profile was too high for a regulatory authority. However, in spite of this view, AECB became the focal point of public interest on TMI-2 in Canada. They believe that AECL would have been better equipped to have handled this type of enquiry. In fact AECL kept out of the controversy.

143. A report entitled "Preliminary Evaluation of the Safety Features of Bruce NGS Relative to the TMI-2 Accident" was handed to me.

Canadian Legislation

144. In response to my enquiry on the status of Bill C-14, the proposed new legislation "The Nuclear Control and Administrative Act" which had been tabled in November 1977, I was informed that this Bill was now dead. This was due to the change of government, which proposed to re-examine the whole nuclear scene. The old 1946 Atomic Energy Control Act, as amended, is accordingly still in force. However, the existing Act gives the AECB wide powers, including the regulation of research reactors and uranium mining and milling operations.

Reactors

145. The AECB have resident inspectors on all nuclear power station sites. On the larger plants, Pickering and Bruce, there are four.

146. Since the jurisdiction of the AECB does not extend beyond the boundaries of a nuclear plant, off-site emergency procedures and contingency plans fall within the jurisdiction of the Provincial Authorities. However, because of the interdependence of on-site and off-site emergency procedures, the AECB requires, as a licence requirement, that licensee collaborate with Provincial Authorities in drawing up emergency plans. The AECB provides advice and assistance to Provincial Authorities in drawing up plans.

147. Reference dose limits for accidents are currently under consideration and should be available shortly. They will propose a range of six action levels. (The AAEC should obtain these as soon as possible.)

Uranium Mining and Milling Operations

148. In 1976 a decision was taken by the Board to undertake a more active role in the implementation of the provisions of the Atomic Energy Control Act on mining and processing of ores containing uranium and thorium. Subsequently, detailed procedures and requirements for the licensing and regulation of uranium and thorium mine-mill facilities were promulgated. Copies of the various documents promulgated since the 1976 decision were given to me and are listed in the Appendix. These have now been accessioned into the AAEC Head Office library. Strict regulatory controls are now being exercised over health and safety of miners, waste management, particularly tailings retention systems, and rehabilitation of mines Details of the requirements are contained in the various reference material listed in the Appendix and these should be referred to for further details.

149. No waste rock or tailings can be removed from a facility although they can be used for grading within a facility. Tailings management appears to be in line with Australian philosophy, i.e. store in reasonably impermeable structures,

graded on decommissioning, with particular care to run-off and re-vegetation (Ref. 4).

150. Clean-up of old uranium mining sites has been a matter of concern all across Canada (Refs. 5 and 6).

151. A matter which has caused some concern is the presence of natural radon in soils in the vicinity of uranium mines. Canadians term this "soil gas" and can be of concern in nearby communities for housing mining workers. It appears to be associated with granular soil (not clays). (References to this phenomena are given in ref. 7.)

Discussions with the USNRC, Washington, 9-10 July

Possible TMI-2 repercussions on licensing and regulatory requirements

152. The NRC was conducting an in-depth review of all aspects of its role, functions and technical regulatory requirements following TMI-2. This review was being undertaken by three separate groups:

(i) The Rogovin investigation

To get an independent investigation, on May 14 the NRC named Washington attorney Mitchell Rogavin of Rogavin, Stern and Hugoe, to hear an investigation of events surrounding the accident. Its investigation, which is reputed to cost about $2 million dollars, will concentrate on the organisational and procedural aspects of the NRC's response as well as technical issues. The relative status of possible Rogavin and Presidential Commission recommendations regarding organisational changes to the NRC is not clear. Both of course will be simply recommendations.

(ii) The NRC's Lessons Learned Task Force

The NRC established a "Lessons Learned Task Force", under the direction of Roger Mattson, in the Office of Nuclear Reactor Regulation, to identify and evaluate those safety concerns originating with the accident that require licensing actions. The Task Force looked into seven technical areas covering:

- reactor operations, including operator training and licensing;
- licensee technical qualifications;
- reactor transient and accident analysis;
- licensing requirements for process equipment, instrumentation and controls;
- on-site emergency preparations and procedures;
- the NRC accident response role, capability and management; and
- feedback, evaluation, and utilisation of reactor operating experience.

(iii) <u>The NRC's Inspection and Enforcement Office investigation</u>

The NRC has directed its Office of Inspection and Enforcement to investigate the facts of the accident to evaluate the performance of the licensee as a basis for corrective action or enforcement action. The investigation is headed by Victor Stello, Director of the Office of Inspection and Enforcement.

153. In the subsequent discussion the following points were made:

- Field monitoring sampling points are required as a licensing condition at all sites. TLDs are placed at these sites and these were found extremely useful as population dose estimates. (N.B. The TLDs were changed every 24 hours after the accident.)

- Over 100,000 persons out to some 20 miles voluntarily evacuated themselves, raising the whole question of the use of Corrective Action Guides (corresponding to Emergency Reference Levels in Australia (ERLs)). The philosophy behind the concept of ERLs is that corrective action will not be taken unless the measured exposure levels justify it. The TMI-2 accident did not follow this principle and both the voluntary evacuations and the subsequent recommended evacuation of children and pregnant women was based on the fear of a possible escalation of the accident (hydrogen explosion) and not the measured exposures which were very low. This question obviously will need to be given considerable attention by the AAEC in relation to our own emergency response plans. (In the context of this it is now clear that NRC prediction of a possible damaging hydrogen explosion was in fact groundless.)

- There are no government plans regarding the issue of iodine for thyroid blocking but some government action on this matter is now likely. A problem, as is well-known in the AAEC, is the shelf life of the various iodine components.

Criteria

154. 10 CFR 100, the Siting Criteria, are now under review by a Siting Criteria Review Committee. This Committee was set up before TMI-2 and the implications of the accident on siting may be far reaching. No indications have yet been given, but a greater requirement on remoteness appears inevitable.

155. The question of Class 9 accidents is also under consideration and there is currently a 3 to 2 majority among Commissioners to include Class 9 accident considerations in the licensing process.

156. Risk Criteria are not yet incorporated into the regulations although an individual risk limit of 10^{-7} is currently being used in the Safety Guides.

Licensing of Research Reactors

157. In the US reactors in excess of 10MW thermal power are classed as Test or Power Reactors and are subject to the same safety requirements. Reactors under 10MW are subject to less stringent requirements, depending upon their power level.

158. In the US, HIFAR and the new reactor design study would be viewed as a Test Reactor and subject to 10 CFR 100 Siting Requirements regarding seismic, environmental considerations, etc. Technical Specifications are also now required although for some older plants in the US these were not required

at the time of licensing.

159. In general the USNRC does not define a "life" for a nuclear plant. It would licence a new plant initially for 10 years. Subsequent extension would then depend upon surveillance reports on the state of the plant.

160. The NRC is currently upgrading the emergency procedures for all research reactors. These requirements will be modelled on the Regulatory Guides for power reactors.

Third Party Liability for Nuclear Damage - Price Anderson

161. $1.2 million dollars had been paid out by Private Insurance Pools to persons evacuated as a result of TMI-2. This had been prompt and on the basis of the initiative of the insurer. However, the broad question whether the TMI-2 accident ranked as an "Extraordinary Nuclear Occurrence" (ENO) and hence whether the provisions of Price Anderson indemnity apply have not yet been resolved. A special review committee has been set up within the NRC to determine this issue; The issues are complex and the definition of an ENO are dependent upon exposure levels far higher than those actually received at TMI-2.

Uranium Mine/Mill Tailings Management

162. It is now recognised in the US that strict licensing procedures are essential for the management of mill tailings. Mining wastes are not regulated by NRC; their control is the responsibility of the States. However, the NRC has found some States without staff of sufficient technical expertise to evaluate the environmental report.

163. NRC regulatory rules for tailings management are now ready for promulgation for public comment. However, as tailings management is heavily site specific, regulatory requirements will concentrate on broad design criteria and performance objectives. A Regulatory Guide on tailings dam design has already been promulgated (Reg. Guide 3.11). Several NRC staff technical position papers were made available (refs. 8 and 9).

Visit to Three Mile Island Nuclear Plant

164. A short two-hour tour of the facility was arranged by

the NRC on 11 July. Major observations were as follows:

Plant Status

165. The plant was in the cold shut-down condition at the time of the visit:

Pressure	325 psig
Primary system water solid natural circulation cooling	IN 150°F
	OUT 159°F
Maximum in-core temp	270°F
Containment pressure	- 0.2 psig
Power level	about 2.5 MW thermal
Heat sink	- one steam generator

General Impressions

166. NRC inspectors were very much in evidence and one gained the impression that utility engineers were "under orders". Even allowing for the disruption that must have occurred as a result of the accident, the general degree of plant order was still untidy.

167. The Control Room appeared to be very dated in design and poor in instrument display characteristic. Instruments which were critical in the accident assessment, e.g. the pressuriser water level and inlet and outlet (cold and hot leg) temperatures etc., were not easily read. However, the major criticism would be the inadequacy of critical plant parameters on display, such as in-core thermocouple readings. Some 50 in-core thermocouples were installed in TMI-2 but these were not calibrated for temperatures above 700°F (371°C) and were fed to the process computer printout. None were on direct visual display in the Control Room (this has now been rectified).

168. I was also surprised to learn that respirators had to be worn at one stage of the incident because of the possibility of high iodine concentrations. In my view, if the Control Room has to be used during incidents it should be heavily shielded and have properly filtered air.

Rehabilitation

169. Plans for cleaning up the plant were under way but a

great deal more planning work is required. At the time of the visit a decontamination clean-up plant was being erected adjacent to the plant-room . The programme is to clean up the 250,000 gallons of contaminated water in the plant room first before proceeding to the more difficult problem of the containment (which contains about a half million gallons of contaminated water (7 ft. from the floor). The clean-up is based on an Epicone 11 demineraliser which contains cation resins in shielded cartridges through which the contaminated water will be pumped. The aim is to dump the freshly decontaminated water into the Susquehanna River. However, injunctions have been placed on the utility against discharging any wastes (even those within the discharge authorisation): and the dumping may not be permitted.

170. Environmental surveys were being conducted by both the utility consultants and the NRC. I asked the NRC representatives how this operation was justified and was told they were simply carrying out an audit.

APPENDIX

LIST OF REFERENCE MATERIAL ACQUIRED DURING VISIT

(Accessioned in Head Office Library)

1. AECB. Guide to the Licensing of Uranium and Thorium Mine/ Mill Facilities. Licensing Guide No. 31, August 1976, Draft.

2. AECB. Uranium Miners Dose Records and Protocol for inclusion in the National Dose Registry. December 1977.

3. AECB. Proposed Regulation for Maximum Permissible Exposure to Radon Daughters. April 1977.

4. AECB. Facilities for Retention of Uranium Mine-Mill Easte. Licensing Document No. 23D. Draft October 1977.

5. Report on Investigation an Implementation of Remedial Measures for the Reduction of Radioactivity found in Bancroft, Ontario, and its Environs. February 1979.

6. AECB. Criteria for Radioactivity Clean-up in Canada. Information Bulletin 77-2. April 1977.

7. Workshop on Radon and Radon Daughters in Urban Communities Associated with Uranium Mining and Processing. March 1979. 2 Vols.

8. NRC. Staff Technical Position Paper. Bioassay at Uranium Mills. June 1978.

9. NRC. Staff Technical Position Paper. Interim Land Clean-up Criteria for Decommissioning Uranium Mill Sites. May 1978.