ATOMIC ENERGY OF CANADA LIMITED
Power Projects, Sheridan Park, Ontario

Lecture 12

# Nuclear power symposium

# LICENSING

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Power Projects

NUCLEAR POWER SYMPOSIUM

# LECTURE NO. 12: LICENSING

by

L. Pease

## 1. INTRODUCTION

I propose in this paper to collect together assorted material related generally to the licensing process, to make some broad generalizations on the matters that a prospective nuclear plant owner should look for, and to give you some leads on "where to dig" for further information. I will be covering the several broad phases involved in the realization of a nuclear project: design, construction, and operation.

I had hoped also to be able to give you some estimate of the size of the effort involved in the licensing process. I found, however, as I got into it, that this is a much bigger job than I had expected, mainly because of the manifold ramifications of licensing, especially of the safety aspects. This affects the design of our plants in such a fundamental way, that it is impractical to put a cost on the plant as it would be if nuclear safety were not involved. It is, nevertheless, something that I would like to do, but it will have to await a later revision of this paper.

However, with the above warning, let me note some miscellaneous facts:

(1) The Atomic Energy Control Board has some 30 full time professional staff, and a total salary budget of $800,000 for the year 1971-72. This is up some $150,000 from the preceding year, and is a reflection of the rising work load of reactor applications and of international activities (chiefly reactor safeguards work in connection with the Non-Proliferation Treaty (NPT)). The AECB has also a sizeable budget which is used in support of research activities in the atomic energy field, a field which includes not only nuclear reactors but also accelerators, radiography equipment, and the like. The Board expended some $8,000,000 in 1970-71, and $11,000,000 in 1971-72.

(2) In addition to the staff in the employ of the AECB, several government departments, federal, provincial and municipal, contribute staff to several advisory committees. The largest is, of course, the Reactor Safety Advisory Committee (RSAC); but the committees on accelerator safety and reactor operators examination, to name two, have a significant work load. There are some 30 people contributed in this way. Some of the committees, especially the RSAC, meet several times per year, and this work is a significant fraction of the work load of some of the staff involved.

(3) The various design departments involved in the design of a nuclear plant, expend a significant effort in safety-related work. The greater share of this work load falls on the reactor design departments, not unexpectedly, because this is where the nuclear part of the nuclear plant is. In safety analysis alone, I have some 15 people involved full time covering all projects. I would hazard a guess that over the course of the design, construction and commissioning of the station, we invest about 10 man-years in the production of the safety analysis. The principal end of this labour is Volume II of the Safety Report. Volume I of the Safety Report is the design description. This is written by the various design branches involved. Apart from the writing itself, how much of this represents work that would not otherwise have been done in any case (for the Design Manuals, for example), can only be conjecture. I would estimate perhaps 5 man-years. These estimates are approximate and can vary widely from project to project. For example, on the first unit of the Gentilly plant, we spent 10 man-years on the analysis of reactor runaway alone.

Safety is big business in the nuclear power plant enterprise, running to several millions on major projects, and appears ultimately in the unit energy cost. It is my personal opinion that safety in the nuclear business contributes a larger share to the product cost than in any other enterprise. However, until someone searches out the facts, it will have to remain a personal opinion.

## 2. LICENSING AGENCIES

It goes without saying that these are government bodies, and that the most important one is the Atomic Energy Control Board.

The Atomic Energy Control Act of 1946 governs all atomic energy matters in Canada. It is specified clearly in the Act that it is to govern the development and control of atomic energy. This Act provides explicitly for the creation of the Atomic Energy Control Board. It is named in the Act, and it is empowered to make regulations, and (through the NRC) grants for atomic energy research. Further, it has complete authority over "prescribed materials". These include, not unexpectedly, uranium and thorium, but also all radioactive isotopes (in excess of "prescribed" amounts) as well as special materials required for the exploitation of atomic energy. Heavy water is one such material.

The Act also authorizes the Minister in charge to incorporate "one or more companies" for the research and exploitation of atomic energy. Two such crown corporations have been formed, Atomic Energy of Canada Limited and Eldorado Nuclear Limited.

It is interesting to note that neither the AECB, nor any company formed under the Act, are immune to law suits or other legal actions in "any court that would have jurisdiction if the (Board, company) were not an agent of Her Majesty".

The Act empowers the AECB to issue regulations, hire staff, etc., in order to carry out its duties under the Act. These are issued from time-to-time. Nuclear power plants are covered by Statutory Order and Regulation (SOR)/60-119, issued in 1960. You won't find the term nuclear reactor in these regulations; you will find rather "prescribed equipment", which by virtue of another regulation includes by definition nuclear reactors. Part VI of these regulations are entitled "Health and Safety Precautions" and contain the permissible exposure levels. In this the AECB follows substantially the recommendations of the International Commission on Radiological Protection (ICRP).

I might note, incidentally, that SOR/57-145, which defines nuclear reactors to be prescribed equipment, excludes specifically federal government reactors. In strict fact, as written, this would include both Douglas Point and Gentilly-1, although in practice these reactors have not been so excluded.

The most important committee of the AECB is the Reactor Safety Advisory Committee (RSAC). A list of members at the time of writing may be found in Appendix 1. The Board has in this way co-opted specialists, all of long experience, in reactor operation, nuclear medicine, reactor control, and metallurgy. In addition to these specialists, the provincial departments of Health, Labour, Environment, and so on, are represented by officers from their staff. These members in general represent but do not act for their department.

This Committee is convened in respect of all applications for license of a reactor facility (save wholly-owned government plants).

The provincial agencies that are involved in licensing include departments of health, labour, and environment, with no doubt provincial variations. If the B. C. departments have different names than their opposite members in Ontario, I hope you can do the translation.

In Ontario, the Departments of Health, Environment, Labour, and the Ministry of Consumer and Commercial Relations (MCCR), have regulatory functions over industry in general, and nuclear in particular.

The Department of Health is responsible for radiation exposure, and, insofar as exposure from nuclear plants is concerned, accepts AECB exposure criteria. (I don't know what would happen if they didn't. A constitutional crsis?) Exposure from other sources would seem to be exclusively a provincial matter, but if a "prescribed equipment" or "prescribed materials" in excess of "prescribed quantities" are involved, the Atomic Energy Control Act is broad enough to give jurisdiction to the AECB.

In practice, of course, the AECB operates by consensus, so that the question of conflict of jurisdiction doesn't arise. Technically, the AECB licenses nuclear power plants, but they probably couldn't be operated if the provincial Department of Health objected.

The Department of Environment administers the Environmental Protection Act (1971), through Water Resources Commission and the Air Management Branch. On the nuclear side these bodies accept the AECB standards, but for other effluents the plants must meet provincial standards. The most significant "other" effluent from nuclear plants is warm condenser water. Ontario Hydro has carried out extensive studies on this for its fossil plants as well as its nuclear. Thermal discharges are not a problem on the Great Lakes, but attention must be paid to the design of discharge structures.

The administration of the Boilers and Pressure Vessels Act (1962/63) comes under the Technical Standards Division of the MCCR (formerly within the Department of Labour). The AECB requires the applicant to get his vessels licensed by the MCCR as a condition for his operating license. The ministry is represented on the RSAC and on the Reactor Operators Examination Committee.

At the municipal level, the applicant must obtain building and business permits. If these should happen to conflict with federal standards, the superior government takes precedence. However, there has never in practice been any difficulty. Another significant contact at the municipal level is the county Medical Officer of Health, the Police, and the Fire Department. The operator is obliged to organize an Emergency Plan in conjunction with these bodies, and to review it annually. This may involve a simulated test, but usually involves updating names, telephone numbers, and the like. The MOH sits on the RSAC for plants in his geographic area.

## 3. CODES AND REGULATIONS

Were it not for the "nuclear" aspects, possibilities of "radiation exposure", and the like, the various vessel, piping and building codes would apply just as to any other enterprise. Indeed, the nuclear codes do not set aside any other codes, but are an addition to them. The nuclear codes have been devised in recognition of the special nature of the radioactive materials being handled. Apart from the possibility (rather remote) of "nuclear excursions", the by-products produced, radioactive nuclei, cannot be degraded or neutralized by familiar chemical processes, because the latter involve the orbital electrons not the nucleus itself. As a consequence, one cannot do anything about these materials save "contain" them. This means isolation in storage locations for as long as they remain radioactive.

This is a long time for some of them, plutonium-239 being the favourite example of many protest factions. Some people have a mental block on this issue, but there is not, in fact, any intrinsic impossibility in the storage of radioactive materials. I wouldn't be at all surprised if future generations devised clever uses for the waste heat and radiation so produced.

However, I am digressing. In this section I propose to discuss the broad features of the codes of design that are specifically "nuclear", under the headings Exposure, Manufacturing and Construction, and In-Service Inspection.

## 3.1 Exposure

The AECB, as noted above, uses the ICRP recommendations on dose and radiological effects. This Committee was created in 1929, and draws its membership from the ranks of the professions in medicine, radiology, genetics, and the like. Its recommendations and studies have been published from time to time in ICRP reports. A partial list of these by number and title is given in Appendix II. The broad field of dosimetry is, of course, intimately involved here, dealing as it does with the estimation of radiation dose as a consequence of exposure, whether by external sources or whether by ingestion of radioactive materials into the body. I don't think that any of you want to dig deeply into this field, but I would recommend Peter Barry's report, AECL-1624, for an account of the exposure/dose relationships of the radioisotopes most of interest to the nuclear power plant operation.

The dose limits to the public which are accepted by the AECB, are given in Appendix A to Don Hurst's CNA paper. The safety criteria adopted by the AECB are found in Appendix B of the same paper. I have reproduced this as Appendix III of this report for your convenience.

## 3.2 Design Manufacturing and Construction

Insofar as existing codes are applicable, they must be applied. Thus, for example, if the Factories Act (or some equivalent) requires that stairs be enclosed (for operator safety), this must be observed even though not covered in any nuclear code.

In Canada, the design, manufacture and inspection of vessels is governed usually by a provincial boilers and pressure vessels act.

These acts generally refer to certain publications of the Canadian Standards Association, American Standards Association and the American Society of Mechanical Engineers, and imply that the rules of these publications shall be followed.

CSA Standard B.51 - Code for the Construction and Inspection of Boilers and Pressure Vessels, and the ASME Boiler and Pressure Vessel Code are the most important ones.

These codes were originally written to ensure a satisfactory level of performance and reliability of boilers and pressure vessels. They are similar to the codes developed and used in other countries.

In early CANDU reactors, vessels were designed and built to either Section I, Power Boilers, or Section VIII, Unfired Pressure Vessels of the ASME Code. The specification ASA B31.1 was used for piping.

As the design of nuclear power plants developed, it became evident that these non-nuclear codes were inadequate and that new codes for nuclear use were needed.

Consequently, in the middle 60's two new codes were issued for nuclear systems and components -- Section III of the ASME Code for Nuclear Vessels, and ASA B31.7 for Nuclear Piping.

Subsequently Section III of the ASME was amended to include piping requirements and renamed Nuclear Power Plant Components.

As experience was gained in the operation of nuclear power plants, the AEC in the United States expressed the need for an in-service inspection program. This resulted in the publication in 1970 of Section XI of the ASME Code - Rules for In-Service Inspection of Nuclear Reactors Coolant Systems. This code is concerned only with components whose failure could affect the public health and safety. It is written for light water cooled and moderated reactors, and therefore is not directly applicable to CANDU type reactors.

At present there is no published code for in-service inspection of CANDU type reactors. However, extensive discussions have taken place between the regulatory authorities, plant designers and station owners and operators to develop in-service inspection programs for the presently operating CANDU reactors.

Eventually it is hoped that a code for CANDU type reactors will be written and approved.

Codes issued by other national bodies, such as the American Society for Testing Materials (ASTM), and the American National Standards Institute (ANSI), formerly ASA, are frequently used. Generally, these codes are not mandatory, but may be specified to assist in establishing desired quality standards.

As far as the containment structures (reactor building, pressure relief duct, vacuum building, and their penetrations) are concerned, the various codes of practice and building regulations are to be applied insofar as they are relevant. However, none of these cover nuclear applications, and this leaves the civil designer pretty much on his own.

The American Concrete Institute has in preparation a design guide (ACI-349), which will cover American reactors. As a service to the industry, AECL has issued a series of internal reports which summarize AECL practice insofar as a design approach to meeting the AECB siting criteria are concerned. Civil designers may, and in fact Ontario Hydro designers do, design still more substantial structures than are necessary strictly to meet the criteria. I might note here that the basic safety criteria "defense in depth" (process, shutdown and containment) is evidently aimed at failures in the process system. Forces on the process system, the reactor itself, and the containment structures from external sources (earthquakes) render a defense in depth approach impossible -- the forces affect both the process and containment system simultaneously. The defense in this case is simply that the process, safety, and safety systems, and their supports must take the combined loads from earthquake forces and process failures simultaneously.

## 3.3 In-Service Inspection

John Sainsbury's lecture on Accident Analysis covered the "defense in depth" philosophy which underlies the siting criteria. Obviously it is a good thing from the point of view of safety to design defense mechanisms into a plant to cover unforeseen events of many kinds. That is the point behind independent braking systems in some modern cars. It is not a bad idea, nonetheless, to have a look at the system once in a while to see whether it shows signs of deterioration, and whether it is capable of carrying out its intended function. That is to say, "prevention is better than cure". Broadly, this is the point of in-service inspection.

I don't think I have the time, and I don't think this is the place to go into detail on in-service inspection. I hope, at the same time, that I do not oversimplify this subject.

In-service inspection begins with the design of the plant, because the places to be inspected must obviously be accessible. However, the people who carry out the inspection are the owners. The inspection may or may not be done by staff in the employ of the plant owners, but whether the owner uses his own staff or whether he hires the service, cost and results are the object. I would expect, therefore, the owner to take a rather active interest in this, and I do not think it amiss to spend rather more time on this.

I propose to discuss this subject as to its function (why it's done), systems to be inspected (what it is supposed to do), and techniques (how it's done).

The reason for in-service inspection may seem to you self-evident, it certainly does to me, but let me go over it. I think it common experience that after the purchaser and user of a piece of equipment (whether it be a household appliance or a turbo-alternator) is satisfied that he has received it in a good state, and that it is safe to use and will indeed perform its function within the limits intended, he thereafter keeps on the lookout for signs of deterioration. There are three reasons why he does this: sudden failure may be an economic loss, or a hazard to the operating staff, or a hazard to the neighbourhood. The important point, however, is that the owner should not put it in service in the first place until he is satisfied that the device can be operated reliably and safely. If this were not the case, obviously the manufacturing inspection should be tightened.

The purpose of in-service inspection is to monitor the plant for signs of deterioration.

I apologize again for labouring this point, but you would be surprised at the number of people who fail to make this distinction, and think of in-service inspection as a means to detect the flaws that the manufacturing inspection missed.

Prevention is the name of the game, but to prevent what? All kinds of things can happen in a large complex plant. These range from messy spills which may be costly to clean up, through process accidents which may endanger the operating staff, to major system breakdowns which may pose some hazard to the neighbourhood. The owner may indeed want to institute inspection or preventive maintenance routines to avoid mishaps which have purely economic consequences, and such are not the concern of in-service inspection. Accidents which may endanger the operating staff and the inspection and safety provisions therefor, are the traditional responsibility of the provincial department of labour, by whatever name it is known these days (in Ontario it has been renamed recently to the Ministry of Consumer and Commercial Relations). This Ministry enforces the Industrial Standards Act and the regulations of the Workmen's Compensation Board.

Equipment failures, however, which may pose some threat to the neighbourhood, range from minor radioactive spills, which are a cleanup problem and which may interrupt the operation of the plant, but which are contained by the containment, to major ruptures in the secondary heat transport system, against which an effective containment may be prohibitively expensive. This would be the case, for example, with the steam drums. This part of the system does not carry a radioactive fluid. Consequently, release of the fluid itself is of no concern. However, the forces involved in, say, a hypothetical circumferential fracture, are so large that effective restraint is impractical. Of course, such a rupture is extremely improbable, but its consequences are only conjectural.

This provides the broad basis for choice of equipment to be inspected: basically equipment which, should it fail, gives rise to forces, pressures, consequential damage, and the like, which cannot clearly be foreseen, and for which the capability of the containment may be difficult to assess. The steam drums already mentioned are one such example, pump flywheels, reactor inlet/outlet headers, and the like, are other examples. On the other hand, small piping, feeder pipes being one such example, are clearly containable by the containment, and are not subject to in-service inspection.

As to the techniques of in-service inspection, these are limited to what can be done from the outside. (The interior of the steam drums may be an exception. These can be inspected by the traditional means from the inside, but the radiation field would make such extensive inspection impractical.) Visual inspection of the outside surface, and ultrasonic soundings are the only techniques available at the present time. Ultrasonic soundings can be used to monitor the progress of sub-surface flaws, and to monitor the change in thickness of the vessel (by reason of corrosion). The corrosion process may itself, of course, be monitored by the use of corrosion coupons of the same material as the vessel and exposed to the same conditions of coolant chemistry, temperature, and the like.

There are other techniques which are in the laboratory at the present time. These are acoustic emission and interference holography. Acoustic emission depends on the fact that as a crack enlarges, the slippage of the grains emits ultrasonic noise. Cracks which are on the point of becoming self-propagating are very prolific sources of such noise. There are, of course, all sorts of minor cracks and imperfections in any structure, and all of these will emit noise. The method will be useful provided it can be shown that an incipient running crack drowns out all the competing sources.

Interference holography is a novel application of a laser light source. It depends basically on the wave nature of the light emitted by a laser, and on the fact that it is a coherent source of light. One can use it to detect, in principle, motion of an object to a resolution of the order 1/4 the wave length of the light being used. In principle, one would use the device to monitor the deformation of the object in question under a change in load. Anomalous fringe patterns would indicate the presence of material inhomogeneities. Whether these are cracks, or mere changes in metallurgical structure due to, say, the welding process, is the crucial question. Even after resolving this it will be necessary to develop the laboratory techniques substantially before they can be used in the field. For example, gross motion of the object (building or equipment vibration) must be prevented, and in the laboratory this is accomplished by mounting the apparatus and the object on a heavy pneumatically supported slab. Evidently a rather difficult feat when one is talking about a large heat exchanger or pump bowl.

I might note finally, that I have omitted entirely the question of inspection of the safety systems. You may recall that the siting criteria require that these systems have a demonstrated unavailability not greater than $10^{-3}$ yr/yr. It is the function of the operational testing program to demonstrate that this target is in fact met. Such testing is in a real sense of the term "in-service" inspection, but people in the nuclear business use the term in a more restrictive sense.

## 4. THE LICENSING PROCESS

In this section I propose to summarize briefly the material presented by Hurst and Boyd in their paper to the CNA of May 1972. (As noted earlier, I have included it for convenience in Appendix III of this report.)

The first stage in the licensing process is site approval. This is not strictly a formal stage, and does not require the convening of the RSAC. The discussions are held with the AECB staff, and are intended to identify special requirements (if any) that the Board may foresee in the use of the site proposed. In Bruce, for example, the Board expressed a concern for the construction staff arising from the operation of the heavy water plant, and in fact the reactor plant was relocated on the site.

The next stage in the process is the application for a construction license. This requires the submission of a preliminary safety analysis report (PSAR) in sufficient detail to show the probable ability of the plant to meet the siting criteria, especially the defense in depth and independence of the safety systems. The permission granted at this stage is usually qualified: that is, construction may proceed to, for example, the point of installing nuclear equipment. The reactor vessel is usually the first such equipment. At this stage the owner must present further details of the design and certain accident analyses as specified by the Board. The construction licensing proceeds stepwise in this way during the construction process. The PSAR is updated annually, and a formal presentation made to the RSAC. Of course, the AECB has one or more staff members appointed full time to observe the progress of the reactor design, and to inform the Board and the RSAC. The terms of the construction license allow the owner to proceed as far as stage C in the commissioning process (the hot testing).

At this point permission to load fuel is requested of the RSAC. Several documents are required at this stage. A final safety analysis report (FSAR), a full set of operating manuals, a site emergency plan, and a full complement of licensed operators. The Board then generally gives permission to load fuel and to commission to stage B, first criticality. On the basis of results from this stage approval to go to full power is then given.

The final stage in the licensing process is the application for an operating license. This is granted after the successful commissioning and debugging of the plant. It is subject to annual renewal, and requires an annual report from the station. The report must cover, amongst other things, unusual incidents, unsafe failures, safety system test results, activity releases, and results of environmental monitoring. This subject is covered in Bob Simmons' lecture (No. 10), and I will say no more about it.

## 5. COMPARISON OF U. S. AND CANADIAN PRACTICE

I can touch only on the highlights of this subject, simply because I am not an expert in the safety and siting of light water reactors.

As far as the broad siting criteria release limits, dose limits, quality assurance provisions, and the like, are concerned, I would have to say that there are more similarities than differences between pressure tube reactors (Canadian) and pressure vessel reactors (American). However, I will note a few of the more important differences. The pressure vessels and the heat exchanger shell (of PWR's) are assumed inviolable. I might note here that the inviolability of the pressure vessel extends to the so-called "safe-ends" of the external pipe connections. Although not explicitly stated, the fabrication and inspection requirements of the vessel and its safe-ends are sufficiently extensive that in the opinion of the licensing authority the risk of failure is negligibly small.

A second significant difference is that the light water reactor design criteria make a negative void coefficient a mandatory requirement (10 CFR 50, Appendix A, "General Design Criteria"). No light water reactor can be built or operated at the present time with a positive void coefficient. (I might note here as a matter of interest, that a negative void coefficient is not pure gain on the safety side. In BWR's, for example, a sudden closure of the turbine stop valve provokes a rise in system pressure, a collapse of the voids in the core, a positive reactivity transient, and reactor excursion as a consequence. The safety of the plant in this case depends absolutely on the shutdown system.)

The U.S. criteria are embodied in the American Codes of Federal Regulations (CFR). Title 10 of these regulations deal with nuclear energy. There are several chapters (parts) within this section of the regulations, some of the more important are:

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| 10 CFR Part 20: | Standards for Protection Against Radiation |
| 10 CFR Part 50: | Licensing of Production and Utilization Facilities (covers reactor design) |
| 10 CFR Part 100: | Reactor Site Criteria. |

The mode of operation of the AEC in conduct of a licensing process is, up to a point, rather similar to the Canadian AECB. The AEC appoints, for example, an Advisory Committee on Reactor Safeguards (ACRS) for the same purpose as the RSAC. It goes about its business behind closed doors. This Committee makes its recommendations to the Atomic Safety Licensing Review Board (ASLRB). This Board issues or withholds the license, and in this respect is like the AECB, but unlike the AECB it issues a notice of intent to grant a license, and convenes a public hearing to receive comment and (usually) objections from interested parties. The effect that objections and injunctions can have on a utility at the operating license stage are devastating. They are extremely vulnerable at this point, because the cost of delays in the operation of the plant is very high. There are many examples in the U.S. of utilities which have found it cheaper to give in than to fight on a matter of principle. It is also rather curious that the private utilities are the only ones that have been hurt this way. The Tennessee Valley Authority, for example, has been free of this trouble. Why, I do not know.

A further organizational difference between U.S. and Canada resides in the fact that licensing and regulation in Canada are the responsibility of the AECB. Promotion and development, on the other hand, is the responsibility of AECL. Both organizations report, of course, to the same Minister, who is responsible in turn to the government. In the U.S., both of these functions are carried out by the AEC under the Chairman of the Commission, who reports in turn to the Joint Committee on Atomic Energy, a committee of the American Congress. This seems in some respects rather similar to the situation in Canada, the AEC Chairman occupying somewhat the same role is the responsible Minister in Canada, but to American protest groups, at least, licensing and promotion appear to be under the same roof. It's a mighty big roof, of course, and whether in fact the development activities of the Commission influence significantly the licensing activities is hard to say. The American critics think so, and point by way of example to the slowness of the Commission to carry out the emergency core cooling tests. In Canada, our critics have pointed out that the AECB and AECL have some directors on their Boards in common. This, of course, is perfectly true, but the responsible Minister does receive his advice on licensing on the one hand, and promotion on the other from separate Boards rather than a single individual, and some at least of these directors do not have a vested interest in atomic energy.

A review of the environmental uproar, the reactor safety issue, would make fascinating reading. I do not have the space here (even if I had the time to do the necessary research) to do more than describe briefly some of the more significant controversies.

Probably the greatest of these is the Calvert Cliffs case. This plant is located on the shores of the Chespeake Estuary, and was intervened against in 1970 on account of the effect of thermal discharges. The USAEC disclaimed responsibility for these discharges, claiming that these were the responsibility of the state agencies. The intervenors took this to the Supreme Court, which decided that the AEC was indeed responsible for reviewing all aspects of the effect on the environment. This meant that the Commission was obliged to demand an environmental impact statement from the licensees, and this requirement in effect put a moratorium of several months on the processing of license applications. There were at that time 66 applications involving 97 reactors before the ASLRB.

The Congress provided some relief to utilities for which delay of their nuclear plant would put them critically short of generating capacity. This came in the form of legislative permission to the AEC to issue interim operating licenses while awaiting the EIS.

Even this was enjoined against in the case of Quad Cities (Commonwealth-Edison and Iowa-Illinois Gas and Electric). The suit was dropped when the owners agreed to a $30,000,000 cooling canal.

The issuance of licenses came to a halt in early 1971 for 17 months until May 1972. Since that time five operating licenses and five construction licenses have been issued.

Some utility spokesmen in the U.S. are predicting ten years or more from site selection to operation, unless legislative relief is granted.

## 6. SUMMARY

### 6.1 The Licensing Authorities

(1) AECB - issues the licenses, but requires agreement of,

(2) Provincial Departments of Health, Labour and Environment, and

(3) Local Councils and MOH.

### 6.2 The Codes and Regulations

(1) Siting Criteria.

(2) Boiler and Pressure Vessel Codes - Nuclear Amendments.

(3) Building Codes - do not cover nuclear applications as yet.

(4) Environmental Protection Regulations.

### 6.3 Licensing Stages

(1) Site Approval.

(2) Construction License.

(3) Operating License.

(4) Annual Review.

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# APPENDIX I

## REACTOR SAFETY ADVISORY COMMITTEE

Members

| | |
|---|---|
| Dr. D. G. Hurst (Chairman) | President, Atomic Energy Control Board. |
| Professor L. Amyot | Director, Institute of Nuclear Engineering, Ecole Polytechnique. |
| Dr. A. K. DasGupta | Acting Chief, Radiation Protection Division, Scientific and Technical Services, National Health and Welfare Department. |
| Mr. G. M. James | General Manager, Plant Administration and Operations, Atomic Energy of Canada Limited. |
| Dr. C. A. Mawson | Head, Environmental Research Branch, Atomic Energy of Canada Limited. |
| Mr. N. S. Spence | Head, Nuclear and Powder Metallurgy Section, Physical Metallurgy Division, Mines Branch, Department of Energy, Mines and Resources. |
| Dr. C. G. Stewart | Chief Medical Officer, Atomic Energy of Canada Limited. |
| Dr. E. G. Letourneau | Clinical Consultant, Radiation Protection Division, Department of National Health and Welfare. |
| Dr. A. Pearson | Assistant Director, Electronics and Reactor Control, Atomic Energy of Canada Limited. |
| Mr. J. H. Jennekens (Secretary) | Scientific Adviser - Reactors, Atomic Energy Control Board. |
| Mr. T. J. Molloy (Assoc. Sec.) | Associate Scientific Adviser, Atomic Energy Control Board. |
| Mr. J. P. Marchildon | Associate Scientific Adviser, Atomic Energy Control Board. |

Members for Ontario Reactor Projects

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| Dr. J.H. Aitken | Chief, Radiation Protection Services, Ontario Ministry of Health. |
| Dr. E. Mastromatteo | Director, Environmental Health Services, Ontario Ministry of Health. |
| Mr. H.A. Clarke | Assistant Director, Division of Industrial Wastes, Ontario Ministry of the Environment. |
| Mr. H.Y. Yoneyama | Technical Standards Division, Ministry of Consumer and Commercial Relations. |

Members for Quebec Reactor Projects

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| Dr. J. Lamoureux | Hôpital Notre Dame de Montréal. |
| Mr. G. Lapointe | Directeur Général, Services Techniques, Ministère du Travail et de la Main-d'Oeuvre. |
| Dr. J.M. Légaré | Division de l'Hygiène du Milieu, Ministère des Affaires Municipales. |
| Mr. G.R. Boucher | Directeur Général, Direction Générale Energie, Ministère des Richesses Naturelles. |

Member for Bruce Nuclear Establishment

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| Dr. D.R. Allen | Director and Medical Officer of Health, Bruce County Health Unit. |

Member for Pickering Project

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| Dr. G.W.O. Moss | Medical Officer of Health, City of Toronto. |

## *APPENDIX II*

## PARTIAL LIST OF ICRP REPORTS

1. ICRP Publication 2: Report of Committee II on Permissible Dose for Internal Radiation (1959).

2. ICRP Publication 3: Report of Committee III on Protection Against X-Rays up to Energies of 3 MeV and Beta and Gamma Rays from Sealed Sources.

3. ICRP Publication 4: Report of Committee IV (1953-9) on Protection Against Electromagnetic Radiation above 3 MeV and Electrons, Neutrons and Protons.

4. ICRP Publication 7: Principles of Environmental Monitoring Related to the Handling of Radioactive Materials.

5. ICRP Publication 8: The Evaluation of Risks from Radiation.

6. ICRP Publication 9: Recommendations of the International Commission on Radiological Protection (Adopted 17 September 1965).

7. ICRP Publication 10: Report of Committee 4 on Evaluation of Radiation Doses to Body Tissues from Internal Contamination Due to Occupational Exposure.

8. ICRP Publication 11: A Review of the Radiosensitivity of the Tissues in Bone.

9. ICRP Publication 12: General Principles of Monitoring for Radiation Protection of Workers.

10. ICRP Publication 14: Radiosensitivity and Spatial Distribution of Dose.

11. ICRP Publication 15: Protection Against Ionizing Radiation from External Sources.

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# APPENDIX III

## REACTOR LICENSING AND SAFETY REQUIREMENTS

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The Atomic Energy Control Board, in its reactor licensing, proceeds through the stages of Site Approval, Construction Licence and Operating Licence. The basic information requirements are outlined in the paper. With increasing experience there have been some evolutionary changes in design and operating requirements, although the radiation dosage criteria remain essentially the same. As an alternative to the conceptual division for safety evaluation into process systems, protective systems, and containment, a nuclear plant may now be regarded as composed of two groupings of process systems and safety systems. The target reliabilities for safety systems have been made somewhat more stringent. Some possible trends in safety criteria and licensing requirements are outlined.

Although considerable attention is given to effluents and to radiation exposures from normal operation, the licensing process will continue to concentrate on ensuring that the chance of a major release of radioactive fission products is *negligibly small.*

### INTRODUCTION

The Atomic Energy Control Act gives the Atomic Energy Control Board broad powers which clearly should be used in the interests of public radiation safety. Accordingly, as the nuclear power program was getting underway, the Board published an order classifying nuclear reactors as "prescribed equipment" under the Act, and establishing the requirement for a licence. Both construction and operating phases are licensed, but at an early stage the applicant is required to provide information on the proposed site and reactor, in effect seeking assurance from the Board and its advisers that they see no fundamental bar to the eventual licensing.

Construction is defined as beginning with the pouring of concrete or erecting of essential foundations for the reactor proper. Issuance of a construction licence implies approval of the general design or design specifications as suitable for the site in question, but it does not mean that an operating licence will automatically be granted. In Canada details of design are normally still under consideration when civil construction begins and these details are kept under review as construction proceeds.

The operating licence authorises operation of a plant within certain defined limits, including the use in the reactor of fuel and heavy water which must be obtained under separate Board orders. Start-up and the early operation are usually covered by an interim operating licence with special conditions and restrictions.

In 1956 the Board created the Reactor Safety Advisory Committee to advise it on the health and safety aspects of nuclear reactors licensed by the Board. This Committee is composed of senior engineers and scientists chosen because of their individual competence, together with technical representatives of relevant federal and provincial departments and local medical officers of health. The representatives vary, depending upon the location of the station. No reactor has been licensed by the Board without first being reviewed and approved by this Committee. The extent and detail of the Committee's review depends, of course, on the complexity, novelty, and size of the project.

The Board staff performs a role supporting and complementary to that of the Committee in the detailed review of design and analysis. It assists the Committee by reviewing the submitted documents and giving advice on technical matters. It also undertakes inspection and compliance reviews at the sites, and approves design and procedural changes within the terms of the licences.

### LICENCE REQUIREMENTS

Although *site approval* is not a formal licensing stage, applicants are encouraged to hold exploratory discussions with the Board staff and the Reactor Safety Advisory Committee when requesting approval of a site. At this time the entire project may be in a very preliminary stage and it is necessary only that the plant size, reactor type, and proposed containment method be identified, together with general information concerning the actual or proposed site or sites.

More detailed information pertaining to the site, such as land use, population, principal sources and movements of water, water usage, meteorological conditions, and geology, is required when a formal request is made for a *Construction Licence*. Technical information on the reactor and auxiliary equipment is also required with the application for a Construction Licence, and this is usually submitted in a comprehensive report sometimes termed a "Safety Report" combining the design description and specifications and the preliminary analyses of accidents. Although many aspects of the design may not be firm, the design description and specifications must provide a clear picture of the plant design and be sufficiently complete to enable independent analyses to be done. The Board has prepared, as a guide for prospective licensees, a document entitled "Requirements for Safety Report".

The granting of a construction licence does not imply acceptance of every argument or conclusion in the Safety Report. The Reactor Safety Advisory Committee and the Board staff, while not accepting the specific claims made for certain aspects of the design, may conclude that they are adequately safe. For example, the report may claim an extremely low unreliability for a component system, whereas the Committee, while not endorsing the value quoted, might accept the system as adequate.

Since many details of the design may be undecided at the time the construction is licensed, subsequent submissions and revisions to the Safety Report are required as the design progresses. The submission and acceptance of such information may be made a necessary condition for carrying the construction beyond a certain stage. In general, the design descriptions and supporting analyses of major reactor systems must be submitted well before these systems are installed. From time to time throughout the period of design and construction the Reactor Safety Advisory Committee and the Board staff meet with the applicants.

The issuing of the *Operating Licence* implies acceptance by the Board of the safety aspects of the plant as constructed. Permission for full operation may be preceded by two substages of authorisation: 1) permission to load fuel; and 2) permission to start up. Prior to loading of fuel, all reactor systems affected by having the fuel in the reactor must have been satisfactorily tested as far as it is possible to do so. The permission to start up requires assurance that all reactor and auxiliary systems have been constructed according to the design and have been satisfactorily commissioned to the extent possible prior to start-up of the reactor. The design description and accident analyses must have been brought fully up-to-date. The operating procedures, the organisation of staff and senior members of the operating staff, must all have been approved, and there must be an approved procedure for handling emergencies involving radiation.

The operating licence includes (either by listing or by reference) conditions and restrictions on the level of radioactive effluents from the plant, the test conditions, and on allowable modifications to the plant and procedures. The Board receives formal annual reports on operation, radiation exposures and radioactive effluents, but the staff reviews these on a continuing basis.

## SAFETY PRINCIPLES AND CRITERIA

### Background

The major hazard, of course, arises from the large inventory of radioactive fission products produced and contained in the fuel. Therefore, all criteria are directed (i) toward minimizing the chance of mechanical failure of the fuel and (ii) to preventing or minimizing the escape of fission products from the plant if fuel failure occurs. The chance of fuel failure depends upon the ability to ensure that the power produced in the fuel and heat removal from the fuel are properly controlled. The escape of fission products can be prevented by ensuring that there are a number of high integrity barriers, the most important of which is the final containment.

In specifying the requirements to be met by the designer and operator a very useful concept was developed in which the nuclear plant was considered to consist of three systems: the process system, the protective system, and the containment system. If these systems are independent of one another, and if each is of a reasonable reliability, the chance of a significant release of radioactive material to the public domain can be kept extremely small.

For the *process system* the aspect of most concern from the safety viewpoint is the frequency of occurrence of faults which could lead to fuel failure, whereas for each of the *protective* and *containment* systems the important parameter is the unreliability defined as the fraction of time during which the system would not perform its intended function.

Progress was only possible in the application of this philosophy when it was made quantitative. The applicants were required to demonstrate that the frequency of occurrence of significant faults in the process system should be less than 1 per three years, and that the unreliability of the protective devices and of the containment divisions should each be less than $10^{-2.5}$.

The International Commission for Radiological Protection (ICRP) recommends that individual members of the public should not be exposed to more than 0.5 rem/yr to the whole body, not including exposure from natural background or medical procedures, and with ancillary recommendations for special cases. By 1965, the concept of the plant as three systems became associated with dose limits. The 0.5 rem/yr was accepted as the limiting dose to an individual at the boundary of the exclusion zone for normal operation, including releases due to failures of the process system alone, i.e. with the protective and containment systems functioning. In addition to the individual dose a limiting population dose of $10^4$ man-rem/yr per site was also imposed. The day-to-day releases must be sufficiently small to allow for consequences of process failures being held within the overall limits.

For the combined failure of a process system and one of the other systems, presumably having a frequency less than once per thousand years, the dose limits were set at 25 rem whole body and 250 rem to the thyroid with a population dose of $10^6$ rem.

In seeking to ensure that postulated limits of unreliability for the protective system would not be exceeded, the designers and the Board's advisers have made use of the instrumentation philosophy which developed from the lessons of the 1952 accident to the NRX reactor at Chalk River. The triplication of shutdown circuits and other systems not only enhances the probability of correct operation when needed, without imposing unnecessary shutdowns, but also permits complete testing during operation. This detects faults and gives information on reliability. The need for well-defined protective circuitry and rigid rules for its maintenance have been fully recognised in the safety philosophy. The protective system must be such that it prevents fuel failure in the event of any reactor regulating system failure and the emergency core cooling system must be capable of limiting the fuel and sheath temperature so that no more than a very small fraction of fuel is likely to fail in the event of the failure of any pipe or vessel in the primary system.

*Recent Developments*

With increasing experience some modifications to the original concept of three simple systems have become desirable. For example, the containment was treated as a single entity whereas it consists of many sub-systems. Also the blanket assumption of complete failure of the reactor shutdown system gave little incentive to the designers to improve beyond what they themselves considered adequate. An approach is being developed, therefore, which treats the various safety systems as somewhat parallel and requires that there be no significant release of radioactive fission products following failure of any one of the safety systems combined with a failure of the process system. One consequence of this approach is the need for analysis of more potential dual accidents than previously, i.e. any conceivable significant failure of the process system must be reviewed in connection with the failure of any of the safety systems to ensure that the resultant release of fission products is acceptable. The basic criterion is the same as before. However, in the face of the larger number of potential combinations and in view of the larger reactors with their larger fission product inventory, the unreliability and failure frequency requirements have been made somewhat more severe. Each safety system is expected to have an unreliability not exceeding $10^{-3}$. The combined frequency of all serious failures of the process system should not exceed one per three years.

This approach accepts and gives credit for a second shutdown system, but only if it is shown that either of the shutdown systems will fully meet the requirements for any serious failure and that they are independent in design and operation and free from any operational connection with any of the process systems including the regulating systems.

Where the proper operation or effectiveness of a safety system requires the sequential or simultaneous operation of several sub-components, combined failure of these components shall be examined also and may require that they individually meet a more stringent reliability requirement so that the overall reliability requirement of the systems will be met.

Although the limiting rate assumed for serious failure of the process systems may appear high, experience has shown that to achieve it requires a very high standard of quality in large complex plants. To achieve this quality initially and to maintain it during routine operation demands a special effort, particularly for the primary system which is of central importance to safety. The ASME Nuclear Components Code with certain specific exceptions has been applied for several years by the Board in co-operation with the Ontario and Quebec departments of labour. The ASME Code on In-service Inspection of Nuclear Reactor Coolant Systems is being used as far as practicable with full realisation that this code was developed for light-water reactors. It is hoped that the work of the CNA Codes and Standards Committee will soon lead to a modified standard fully applicable to Canadian reactor designs.

The standard of quality necessary throughout a nuclear plant can be achieved best and most certainly through a program of quality assurance that extends from the conceptual design through to operation. The procedures for controlling quality in manufacture are fairly well established but need more rigorous application. However, the concept of quality assurance, through organization, audit, standards, etc., in the design stage is not yet widely accepted or practised. It is hoped and expected that the industry will move fairly quickly in this direction since the requirement for quality to achieve high operating availability parallels the requirement for quality to achieve high reliability for safety.

The standards and principles developed over the past two decades, especially as applied to safety systems, will continue. The requirement to demonstrate physical and functional separation of the safety systems will be, if anything, now more stringent and special design and maintenance techniques may be necessary to ensure meeting it. The passive safety systems must be testable, at whatever frequency is necessary to ensure the required reliability. It will continue to be necessary that the safety systems are effective without unrealistic requirements that could not be maintained in service.

Final reliance for safety of an operating plant lies mostly in the hands of the operating staff. The examination and authorization of key operating personnel continues, and reviews of total staff training, organizational requirements and the role of other personnel in the safety of the plant will be conducted to determine if further controls would be appropriate.

In appendices A and B the criteria and principles are stated more explicitly. Appendix C contains the definition of exclusion zones for nuclear facilities.

### Future Trends

Several of the criteria on which our licensing is based are currently under review and others may be in the near future. The results of these reviews, of course, are difficult to predict with any degree of certainty but the following paragraphs will outline some of the possible directions.

(i) The criteria for man-rem limits, especially those assigned to normal operation, were developed several years ago using available information on the effect of dosage and assuming a linear relation between dose and effect. This subject is under constant review by world authorities such as ICRP, and we shall be guided in our fundamental dose criteria by any modifications in the recommendations.

(ii) Positive void coefficients have been accepted in Canadian power reactors. However, large coefficients impose rather severe demands on the design of the protective shutdown system and accident analysis is then difficult. Future reactors may be required to have a void coefficient within specific limits.

(iii) The need for high quality of the process and safety systems and the growing complexity of the large nuclear power plants is leading to increased emphasis on quality. It is likely that we shall require more organizational control in design and manufacturing of nuclear power plants to oversee, check, and control the safety aspects of the design, procurement, manufacture and installation of important equipment. The quality which is achieved by strict adherence to the pressure vessel codes, the quality assurance programs and the in-service inspection programs will permit an assessment of improved reliability.

(iv) Local investigations may be required to demonstrate the claimed dispersion factors for atmospheric releases and for waterborne releases. While those being used today are believed to be conservative, we may require greater assurance that releases are adding only a small additional radiation dosage to the population.

## SUMMARY AND CONCLUSIONS

The Canadian approach to reactor safety, while benefiting from approaches elsewhere, has developed independently. The lesson of the NRX accident and the specific Canadian reactor concept have helped in this distinction. Some of the principles proposed in Canada have been adopted in one form or another elsewhere. These include the basic probability approach, the separation of safety systems from process systems and from one another, the requirement for testing of passive safety systems and the imposition of a limited man-rem population dose as a design and operating criterion. Every effort will be made to keep our standards consistent with the best approach of other countries and with the requirements of the society in which we live. As the industry develops, it will become essential to express and specify in further detail not only the basic safety criteria but also design manufacturing and operating requirements which will give assurance of meeting the basic criteria. To ensure that the requirements can be met in spite of the complexity of large plants being designed and projected for the future will demand strong organizational control throughout the entire industry, from design and specification through to procurement, manufacture, testing and operation.

Within the past few years public concern for safety of nuclear power plants has at least partially shifted from the question of a major disaster to the effects of normal effluents. While these have always been of great concern to the licensing body, the major concern is and has been to ensure that serious accidents do not occur. Additional requirements may be imposed on radioactive effluents but the major effort of the Board's reactor licensing staff and Reactor Safety Advisory Committee will be in clarifying and strengthening the criteria and in ensuring that the design and operation are such that the probability of a significant accident causing widespread harm is truly negligible.

## APPENDIX A

### OPERATING DOSE LIMITS AND REFERENCE DOSE LIMITS FOR ACCIDENT CONDITIONS

| Situation | Assumed Maximum Frequency | Meteorology to be Used in Calculation | Maximum Individual Dose Limits | Maximum Total Population Dose Limits |
|---|---|---|---|---|
| Normal Operation | | Weighted according to effect, i.e. frequency times dose for unit release | 0.5 rem/yr whole body 3 rem/yr to thyroid[a] | $10^4$ man-rem/yr $10^4$ thyroid rem/yr |
| Serious Process Equipment Failure | 1 per 3 years | Either worst weather existing at most 10% of time or Pasquill F condition if local data incomplete | | |
| Process Equipment Failure plus Failure of any Safety System | 1 per $3\times10^3$ years | Either worst weather existing at most 10% of time or Pasquill F condition if local data incomplete | 25 rem whole body 250 rem thyroid[b] | $10^6$ man-rem $10^6$ thyroid-rem |

[a] For other organs use 1/10 ICRP occupational values

[b] For other organs use 5 times ICRP annual occupational dose (tentative)

# APPENDIX B

## Power Reactor Safety Criteria and Principles

1. Design and construction of all components, systems and structures essential to or associated with the reactor shall follow the best applicable code, standard or practice and be confirmed by a system of independent audit.
2. The quality and nature of the process systems essential to the reactor shall be such that the total of all serious failures shall not exceed 1 per 3 years. A serious failure is one that in the absence of protective action would lead to serious fuel failure.
3. Safety systems shall be physically and functionally separate from the process systems and from each other.
4. Each safety system shall be readily testable, as a system, and shall be tested at a frequency to demonstrate that its (time) unreliability is less than $10^{-3}$.
5. Radioactive effluents due to normal operation, including process failures other than serious failures (see #2 above), shall be such that the dose to any individual member of the public affected by the effluents, from all sources, shall not exceed 1/10 of the allowable dose to Atomic Energy Workers and the total dose to the population shall not exceed $10^4$ man-rem/year.
6. The effectiveness of the safety systems shall be such that for any serious process failure the exposure of any individual of the population shall not exceed 500 mrem and of the population at risk, $10^4$ man-rem.
7. For any postulated combination of a (single) process failure and failure of a safety system, the predicted dose to any individual shall not exceed (i) 25 rem, whole body, (ii) 250 rem, thyroid, and to the population, $10^6$ man-rem.
8. In computing doses in 6 and 7 the following assumptions shall be made unless otherwise agreed to:
   (i) meteorological dispersion that is equivalent to Pasquill category F as modified by Bryant[1]
   (ii) conversion factors as given by Beattie[2].

[1] Bryant, P.M. UKAEA report AHSB(RP)R42, 1964.

[2] Beattie, J.R. UKAEA report AHSB(S)R64, 1963.

# APPENDIX C

## EXCLUSION ZONE

### Definition

An Exclusion Zone is an area, specified by the Atomic Energy Control Board, immediately surrounding a nuclear facility and under the control of the licensee or the operator.

### Conditions

1. There shall be no permanent habitation within the Exclusion Zone.
2. Use of the land for purposes other than the licensed activities shall require separate AECB approval.
3. Exclusion Zones shall be posted in a manner acceptable to the Board.
4. Radiation safety within the Exclusion Zone is the responsibility of the licensee, or, subject to AECB approval, his designate. Methods and measurement for ensuring radiation safety are subject to review as required by the Board.

### NOTE

For all power reactors licensed to date the Exclusion Zones extend from the reactor core to a radius of 3000 feet with the exception of navigable waters and minor other exceptions.