THE APPLICABILITY OF NUCLEAR ENGINEERING SAFETY PRINCIPLES AND CRITERIA TO THERMAL-ELECTRICAL GENERATION STATIONS GENERALLY

by

J.H. Jennekens and A. Wright

Atomic Energy Control Board
Ottawa, Canada

Presented at the
Canadian Electrical Association
Spring Meeting
Toronto, Canada

March 13, 1978
ABSTRACT

The consistently high capacity factors for the four unit, 2,000 MWe Pickering Generating Station provide solid evidence of sound engineering and high quality of component manufacture, construction, maintenance and operation. It is clear that the achievement of this high level of performance is due to a number of contributing factors one of which is the observance of the nuclear safety principles and criteria developed over the last two decades. The application of these principles and of the relevant criteria to all thermal-electric stations would appear to be worthy of serious consideration.

RÉSUMÉ

Les facteurs d'utilisation élevés constamment obtenus pour les quatre tranches de la centrale nucléaire de 2000 MWe de Pickering fournissent une preuve évidente d'une bonne conception et d'une haute qualité des composants manufacturés de même que de la construction, de l'entretien, et de l'exploitation. Il est clair que l'accomplissement de ce haut degré de performance est dû à un certain nombre de facteurs concourants dont l'un est l'observance des principes et critères en matière de sûreté nucléaire développés au cours des deux dernières décennies. L'application de ces principes et des critères judicieux à toutes les centrals thermo-électriques mériterait d'être considérée sérieusement.
INTRODUCTION

There is now worldwide recognition of nuclear power as an economic option for electricity generation, and Canada's nuclear program is firmly established with the success of the CANDU reactor system. In Ontario there are, at the time of writing, seven large operational units (four at Pickering "A" G.S. and 3 at Bruce "A" G.S.) contributing slightly over 4000 MWe to the grid. Also, Canada's first commercial nuclear unit at Douglas Point supplies 40% of its 200 MWe to the grid, with the remainder of its output used for steam heating in the production of heavy water.

Economics have dictated that a nuclear unit should be a base load generator. This is because the capital cost of a CANDU, per unit of installed power, is about twice that for a fossil-fired station, while the running cost per kWh is half in the first year of operation falling to one quarter after fifteen years of operation, of that for the fossil station.\(^1\) The prime requirement for a base load station is, of course, a high availability, so from an economic standpoint, plant reliability is a very important factor in nuclear station design. An additional incentive to avoiding plant outages is a feature peculiar to the nuclear reaction, called "poison-out". On termination of the nuclear reaction, poisons in the core increase above their normal operating levels. After a forced outage, unless the plant can be brought back to power within a short time, usually half an hour, the poison build-up will exceed the positive reactivity. Once this has happened, the plant operators must wait for the negative reactivity effect of certain fission products to peak and then fall off due to radioactive decay. This usually means a period of about one and a half days.

However, experience with the Pickering "A" G.S. indicates that the required high availability has been achieved. Figure 1 shows the performance of the four units over a 2½ year period from January 1975 to June 1977. The important periods to note are the outages and deratings, which are very low for Units 1 and 2. Unit 3 was down from August 1974 to June 1975 and Unit 4 from May 1975 to April 1976, to replace leaking fuel channel pressure tubes. The leaks were through cracks caused by rolling problems with new materials and are confined to these two units. Following the replacement of these pressure tubes, the availability has been excellent.

The obvious question is how has this plant reliability been achieved? Economic incentive has been a powerful spur, but equally important is the spin-off from nuclear safety criteria, and it is this aspect that forms the subject of this paper.
NUCLEAR SAFETY PRINCIPLES

The use of nuclear energy for weapons purposes more than thirty years ago demonstrated very clearly that, in addition to the immediate destructive blast and heat, radioactive materials are produced which can have very harmful effects on people and their environment. These effects included sickness and death as well as environmental pollution.

Although the nuclear reactor in an electricity generating plant is not capable of causing a bomb-type explosion, it does produce the same radioactive material by-products. So from the very outset, those working on the peaceful exploitation of nuclear power were conscious of the need to keep these by-products well away from the public and the environment. Therefore, considerable attention was devoted to radioactive material release prevention. This release prevention effort had two facets: the first was aimed at limiting releases while the plant is running normally (routine emission), the second was concerned with potential releases in the event of an accident to the plant (accidental emissions).

Routine emissions occur along both air and liquid effluent pathways and it is important to limit the quantity of radionuclides in these emissions. This quantity is a function of the integrity and leak-tightness of containing pipes and vessels, and also, air and liquid filters, condensing devices and holding (delay) tanks. The levels of radiation and radioactive material measured in the environment surrounding a power station provide continuous checks on the degree of public protection achieved. Also, performance monitoring of the individual systems within the station gives early warning of a failure.

Protection against the release of radionuclides resulting from accidents to the plant starts with the integrity of the systems needed for normal plant operation—we call them process systems. If they are of high integrity, i.e. conservatively designed with good quality control before and during installation, and adequate routine maintenance, testing and inspection during operation, then the chance of having a process system failure in the first place is minimized. However, if in spite of these considerations, a failure does occur, then its consequences can be reduced by the action of other systems, specially installed for this very purpose. These systems, which we call safety systems, principally aim to terminate the nuclear chain reaction, cool the fuel and contain any radioactive material released as a result of the failure. Obviously, it is very important that such systems be available when called upon and that they function as intended.

The above approach to limiting radioactive releases, both routine and accidental, was adopted over two decades ago by the nuclear plant designers in conjunction with the Atomic Energy Control Board (AECB). Since then, the basic safety principles have been formalized and also amplified. This last aspect is the most significant one and is responsible for the considerable effort and expenditure on nuclear safety in
the last twenty years. It is all very well to require that a system be conservatively designed to achieve high availability; the problem is to break down these general goals into their many facets and then to build in the detailed assurance.

Routine emissions can be controlled in a fairly straightforward manner. At the design stage, the quantities and pathways have to be identified and the appropriate equipment installed. Dehumidifiers remove tritiated water vapour in the reactor building air, and stack filters trap particulates such as $^{137}\text{Cs}$ and $^{88}\text{Rb}$. There are also charcoal beds for adsorbing gases and vapours, chiefly iodine isotopes. All aqueous liquids are collected via the drainage system and pumped to storage tanks for monitoring, and if the radiation levels are sufficiently low, they are pumped out in the condenser cooling water stream. Organic liquids are collected and placed in containers for long term storage or incineration. The effectiveness of the measures is confirmed by the environmental monitoring, referred to previously. The important point is that operational experience is available to check and improve the theory.

Accidental releases, however, present a more formidable problem, since they are essentially the result of events at which considerable effort is aimed to ensure that they will not happen. The absence of such events during commercial nuclear generating station operation is the measure of the success of these efforts. Nevertheless, an excellent track record of prevention is not a guarantee of immunity. We require solid assurance that all will be well even if the unexpected happens.

Therefore, when the preliminary design of a plant's process systems is complete, we identify on a realistic basis all of the postulated ways in which process systems can fail, and choose the worst case for each mode. These are called Design Basis Accidents (DBAs), because they are the bases or specifications to which the safety systems are designed. This means that the designer must assume that a DBA has occurred and show by analysis that the safety systems will limit the radioactive material release to a suitable low value. In this analysis we also require a degree of co-incidental impairment of the pertinent safety systems to be assumed on an individual basis. These process and safety system events then constitute a set of Design Basis Accidents. To be meaningful a DBA must have some probability associated with it. The usual target for any accident for which serious radiological consequences may result is around (and ideally less than) one chance in a million. Accidents which can occur more often should be less severe, releasing less radioactive material.
For this approach to be valid then, we must ensure -

(i) That the DBAs will not occur more often than their target probabilities.

(ii) That an accident resulting in severe radiological consequences has an expected probability of occurrence of less than once in a million reactor operating years.

Accordingly, we have set targets for the performance of systems, and to check on this performance an extensive reporting of faults and incidents during plant operation is required. For process systems, all serious faults are identified. These are those faults which either did call, or could have called, upon the safety systems to prevent an unacceptable release of radioactive material. Our target is that, for every three years of reactor operation, there should be no more than one of these faults. For safety systems the target is that each system must not be unavailable more than about eight hours out of the 8760 hours in a year. This has to be demonstrated by testing.

If the above performance targets are achieved, we can be reasonably certain that any process system failure co-incident with unavailability of a safety system will have a probability of occurrence of less than one in three thousand. The achievement of this low failure rate then gives us confidence that an accident for which the radiological consequences are serious, will have a probability of occurrence of less than one in a million.

One point to note here is the assumption of co-incidental, rather than consequential, safety system unavailability. This is based on a long-standing principle in Canadian nuclear safety, that safety systems should be separate and independent from process systems (and from each other). Total separation is never practically achievable, but we take it further than any other nuclear reactor system and devote considerable effort to ensuring that whatever commonality remains is not a safety problem.

Conditions (i) and (ii) quoted above require a detailed examination of all the plant systems and the postulation of failures and event sequences. Theoretically, there is a very large number of scenarios to be looked at, since a nuclear station has many systems and thousands of individual components. However, since resources are limited, a judgement is made as to which systems or components are crucial to safety, and failure scenarios for these are postulated. We are particularly conscious of "cross-linked" faults. These are faults arising in several systems from a common cause, such as a radio interference pickup at the input of a high-impedance circuit, or the same degraded oil lubricating several pump bearings. Where analysis of such a failure sequence shows a possibility of a significant radioactive material release then its probability of occurrence is assessed. If necessary,
measures are taken to reduce its probability to an acceptably small value. Several areas where the design could be usefully strengthened have been identified by this process, particularly in relation to cross-linked faults. The reporting of process system performance, referred to previously, has proved to be a useful guide when choosing types of initiating faults.

Hopefully the foregoing has given you a reasonable appreciation of our nuclear safety principles and how we apply them. The next section will describe some of the safety practices in more detail and trace the influence of these practices on process system reliability and hence plant availability.

NUCLEAR SAFETY PRACTICE

1. Design Integrity

A sound design is only achieved if attention is paid to the following areas:

(a) Safety Margin

Nuclear components are generally designed with sufficient safety margin to ensure a low intrinsic failure rate. When selecting electrical equipment, conservative estimates are made of the loads and number of cycles expected. The primary heat transport system piping and welds are designed and fabricated to ASME Section III Class 1. This requires design by analysis and includes a safety factor to allow for statistical variation in the properties of materials and variations in dimensions and welding. The material used is ASTM A106 Grade B carbon steel. Where components are subject to cyclic stressing or wear then a pessimistic upper limit to the number of cycles experienced must be defined. Similarly, for corrosion, embrittlement and erosion a conservative estimate is made of the chemical concentrations and flow velocities.

The general experience with the primary pressure-retaining system has been good, as evidenced by the low leakage rate of heavy water. An exception to the good record was the problem of leaking pressure tubes in Pickering "A" G.S. Units 3 and 4. To obtain lower in-service creep rates, these units were fitted with Zr-Nb tubes, but improper rolling to the end-fitting caused the thinner-walled tubes to bulge, and hydriding led to the formation of tiny cracks. However, the moisture detection instruments in the annulus gas system quickly detected the leakage and the tubes were replaced. Also, during the early years of Pickering operation some redevelopment of the fuelling scheme was necessary. Initially fuel bundles, when moved from low to high neutron flux regions in the core swelled and distorted their sheaths, so a new bundle
shift scheme was devised to avoid this steep flux change. Recently fuel elements have incorporated graphite lubricant inside the sheath which permits axial movement of fuel pellets and also absorbs iodine fission products; this should further reduce fuel element failure rate. It should be noted here that the CANDU concept of on-line refuelling is a major factor in plant availability. Light water reactors, which predominate in many countries, require shutdown and removal of the pressure vessel head for refuelling, which involves several weeks downtime.

(b) Quality Assurance

Inadequate fabrication or installation can negate a conservatively designed component or system. Accordingly, the AECP has placed increasing emphasis on quality assurance. The CSA N.286 Standard series is intended to specify the quality assurance systems that a component supplier, installer and operator must establish. These are administrative standards and cover the areas of procurement, design, manufacture, construction and installation, commissioning and operations. Although most of the standards are still in preliminary form their underlying principles have been in use for several years. As an example, manufacturing is covered by the CSA standard Z299.1 which specifies requirements for the control of design, manufacturing, purchasing, incoming inspection, in-process inspection, final inspection, measuring and testing, storage, packaging and shipping, and documentation. The AECP has recently begun to visit manufacturers to determine the status of their Quality Assurance programme with respect to maturity and implementation. We are also drawing up comprehensive lists of priority items based on the importance of components and systems to plant safety.

(c) Design Redundancy

In the event of equipment malfunction, standby equipment may be brought in, either automatically or manually, to take over the duty of failed items. The concept of redundancy is older than the nuclear industry, but the demands of safety have required a much more formal basis for the priority of redundant equipment. Mentioned earlier in this paper was the postulation of event scenarios stemming from failures in process and safety systems. From these the importance of protecting against certain types of process failure has emerged. Of course, many of these are obvious, such as a failed pipe or valve, a stopped pump or an inadvertently withdrawn reactivity control device. This is because there are only three basic modes of dangerous failure in a nuclear reactor: - loss of reactivity control, loss of normal fuel cooling and loss of the normal heat sink. However, the real task is to identify the many causes of component failure and their like-
lihood; which requires reliability studies. From these studies the requirements for redundant equipment (i.e. duplication or triplication of equipment) in some systems have emerged.

Generally, the items requiring duplication or triplication are active components such as pumps, valves, and controllers with their instrumentation; also electrical supply and to some extent service fluid supply. Passive systems or components such as fluid reservoirs are deemed to be sufficiently reliable not to require duplication. Also, it is not practicable to incorporate an alternative full reactor power coolant circuit; instead a shutdown cooling circuit is provided to remove decay heat. In Pickering "A" G.S. there is 33% spare primary coolant pumping capacity; however, the reliability of these pumps has been sufficiently good to justify the removal of pump redundancy on later plants at Bruce and Gentilly.

There are standby pumps and valves on the boiler feedwater systems and, since this system is particularly important for safety support, its total reliability is likely to be improved still further. There is also spare pumping capacity in the reactivity liquid zone control system, and the moderator cooling system at Pickering has 25% spare pumping capacity. Later plant designs at Bruce and Gentilly have 100% spare capacity. In addition to being available in an emergency the redundancy of pumps and valves has meant that routine maintenance can be carried out on the equipment normally in use, thus improving its availability. During a common-cause fault analysis one system in particular was identified as being crucial to safety. This is the service water system which takes water from the lake and cools a large number of auxiliary heat exchangers and pump glands. If this water source is lost then many plant systems would be jeopardized, endangering the core cooling. Consequently an alternative supply was arranged which could function long enough to ensure plant safety. In the event of failure of the normal service water supply a shutdown may occur, but the integrity of many components is maintained.

The integrity of plant electrical supply has also been established as being of high priority as a result of safety considerations. The electric system design follows conventional generating station practice, but the increased reliability required for some nuclear systems has resulted in a more selective bus arrangement, and more standby and redundant equipment. Each of the four units has an odd and even bus division to obtain dual bus security. Loads are connected such that half of the electrical power to any process is supplied from an odd bus and half from an even one. Lower-voltage-level buses have an odd or even designation to match
their source of supply. Auxiliaries, such as the lubrication pump on the Boiler Feed Pump motor, supplied at a lower voltage than the associated primary element, will be connected to an odd or even bus to match the primary element source. The buses are classified into four levels of reliability, the lowest numbered class being the most reliable. Class 4 buses are fed from either the unit or the switchyard transformers and their loads are those which can tolerate a long term shutdown without affecting equipment or personnel safety. Typically they supply the primary coolant circulating pumps and boiler feed pumps. Class 3 buses are normally fed from Class 4 buses but are backed up by standby generators in the station. Class 3 loads are those which can be interrupted for a short time, while the standby generators start up, but are needed for safe plant shutdown. Auxiliary boiler feed pumps, service water cooling pumps, generator hydrogen seal oil cooler pumps, turbine governor valves and emergency stop valve fluid cooler pumps, and service air compressors are typical Class 3 loads. Class 2 buses are a.c. buses fed through inverters or motor-generators from the Class 1 system or from the Class 3 buses in the event of inverter failure. The loads are those which cannot tolerate the short interruptions which can occur for Class 3 power, such as safety-related instrumentation, emergency lighting and power regulating system control valves. Class 1 buses are d.c. buses supplied by rectifiers from Class 3 and backed up by batteries guaranteed to supply the load for approximately 3 - 1 hour. The protective shutdown systems do not need electrical energy to actuate, but their controls are supplied by Class 1 power.

Lastly on redundancy, mention should be made of its application to instrumentation. One of the most important sets of process instrumentation is that for the reactor power regulating system; consequently the measurement of boiler pressure, level, steam and feed flows is at least duplicated, and triplicated in certain designs. Reactor power and core exit temperatures are measured by 20-30 instruments, because of the power distribution, and a comparison and averaging process is used. Experience with shutdown system design and operation dating back to the early 1950's led to the use of triplicated channels with 2 out of 3 signals required for trip actuation. The triplication provided redundancy of equipment whilst the 2 out of 3 logic reduced the spurious trip rate. Also it facilitated maintenance, since the channel being maintained could be set to trip, and then if a dangerous condition developed a trip would still occur if one of the two remaining channels was faulty. Process systems requiring high reliability and maintainability such as reactor power control and boiler-turbine control also incorporate redundancy and fail-safe provisions.
2. In-Service Operation

(a) Monitoring

System monitoring is a means of identifying fault conditions in their early stages so that action can be taken to prevent the occurrence of a major equipment or system failure. Thus monitoring for safety also plays a major role in plant availability, and the nuclear plant has a mass of such instrumentation. Particularly, the primary coolant circuit and its associated sub-systems are provided with instrumentation to verify continued integrity. If process parameters exceed specified limits the reactor is automatically shut down for investigation and repair. The early fuel failures at Pickering NGS, referred to previously, were detected by a fission product monitoring system, and more recently delayed neutron scanning systems have been installed in some plants. Also the leaking primary heat transport pressure tubes were detected while the leak rate was very small by moisture detectors in the annulus gas system. Primary-secondary side leakage would be seen by infra-red detectors, as would any leakage of the moderator heavy water into the service water cooling system.

Generally, however, the value of the monitoring system has been to identify less serious failures so that they can be listed for future attention during a maintenance period. These fall mainly into the small-leakage category; overheating seals and some vibration may also be tolerated for a time. Direct observation is responsible for identifying many failures, but in some areas such as the reactor vault where entry is limited, remote moisture detection instruments are installed at critical points e.g. close to the feeder piping. Leakage through the main coolant pump seals is collected in a vessel with a level or rate indicator. The vault air is monitored for tritium level which would be another indicator of a primary leak. Moisture detectors are also installed in the boiler room to check on secondary fluid leakage. The service water system is sampled for tritium and this provides a useful backup to the infra-red detectors.

The large number of instruments for monitoring involves considerable effort, both in taking the readings and calibrating or maintaining the instruments. However, the fault detection contribution to plant availability is well worth the additional expense.

(b) Inspection, Testing and Maintenance

A prime concern has always been that systems or components important to nuclear safety shall be designed and installed
so as to facilitate inspection. Particular emphasis has been placed on those which are not easily duplicated, e.g. pressure retaining components such as boilers, piping, pumps and valves. These items, together with the methods and frequency of inspection, are listed in CSA N285.4, which is the Canadian Standard for in-service inspection of CANDU nuclear power plant components. A complete inspection of all the listed components and welds is required within a four-year period commencing one year after the generation of power. Subsequent inspections must be completed within intervals of one-third of plant design life or 10 years, whichever is shorter. Circumferential and nozzle welds of selected components are inspected ultrasonically and compared with their pre-operation fingerprints. Piping is inspected visually for surface cracking and volumetric checks are made for possible wall thinning, especially in elbow and tee joints where erosion and corrosion-erosion may occur. A similar visual and volumetric inspection is carried out on pump bowls. The holding studs on pumps and valves are also checked since they are subject to stress cycling and vibration. A considerable number of boiler tubes are inspected, although since performance has been very good, this is not yet an AECB requirement. However, in future it is anticipated that some degree of in-service inspection will be specified, as tube integrity is a vital parameter in Design Basis Accident analysis.

Maintenance is an activity which consumes a considerable amount of effort and expenditure, since the emphasis on redundancy and monitoring has produced a mass of valves, sensing devices, warning indicators, standby pumps and electrical components such as amplifiers, switches and relays. A regular maintenance schedule is essential if the benefits of the additional equipment are to be realized. Testing is an essential part of maintenance and much of this is safety-related. For instance, periodic testing of the standby generators is necessary to demonstrate the required availability of the Class 3 electrical power supply. Also, the triplication of control instrumentation, to reduce spurious response and allow on-line maintenance, means that the number of channels to be maintained is increased.

Most electrical equipment can be tested and maintained while the plant is operating and also many of the valves have isolating valves for this purpose. Turbine governor valves, emergency stop valves and steam intercept valves are just a few examples of where a regular routine of valve-stroking is carried out. However, where plant conditions or needed repairs within the radiation field of the primary circuit do not permit on-line maintenance, then it is done during the planned outage periods. Non-serious faults shown up by the monitoring system are also saved for rectification at these times.
A considerable amount of prior organization goes into a planned outage since every 24 hours that a Pickering G.S. unit is down costs the utility (and later the public) approximately $140,000 in equivalent fossil plant operation. However, economics is not the only consideration— all safety-related maintenance must be completed, and any operations in the radiation field of the primary circuit must be planned for maximum efficiency. This is because the amount of personnel exposure to gamma radiation and tritium must be strictly limited. Often this means that new procedures must be devised to enable the maintenance personnel to spend minimum time inside the reactor building or vault. Occasionally special tools are designed and built for these operations. Another essential practice for efficient maintenance and continuing safety assessment is the recording of all changes to systems during plant life. For example, the utility replaces the design system flowsheets by producing another set called the operational flowsheets which are continually updated during plant life.

(c) Incident Reporting

In compliance with the AECB requirement to demonstrate that the target of no more than one serious process failure per three years of plant operation has been met, there is an extensive system of incident reporting. A wide variety of fault types is reported and a system of classification has evolved to rate the seriousness of each one. Five classes are specified ranging from Type A—a fault that could cause significant fuel failures in the absence of safety system action—down to Type E, which is a failure that either has no effect upon or would tend to lower fuel temperature. Serious process failures fall into the Type A category and the AECB takes special note of these, consulting with the utility to ensure that a repetition of the particular fault and other possible similar ones will not occur.

Sometimes such failures, in addition to their implications for safety, bear upon plant availability. An example of this was provided by the digital computer-controlled regulating system at Pickering G.S. The increasing reactor size made the task of controlling the neutron flux distribution within design limits, very complex. As a result digital computer control was selected over a "hard-wired" analog system, since this would allow greater flexibility to make changes to the control algorithms that might be needed as a result of commissioning and early power operation. However, the new system initially demonstrated a serious failure—and a safe failure—probability significantly higher than the earlier analog systems. This was due both to malfunction of equipment external to the computer—e.g. zone control valve hardwired logic, and also to loopholes in the digital computer
programming, mainly associated with self-checking routines. Because of this, safety standards as well as production availability were being reduced, and a comprehensive review was undertaken. As a result, hardware and programming changes were made and better administrative control over the programming was instituted. More recent experience has shown a marked improvement in the system performance from both the production and the safety aspect. However, we are continuing a close monitoring of the situation through the incident reporting procedures.

CONCLUSION

The principles of conservative design, quality assurance, thorough inspection and maintenance are not new. The extent to which they are carried, however, is the distinguishing feature of nuclear electrical power generation. Economic incentive alone would not have produced the reliability of plant systems that we see typified by the performance of Pickering G.S. today. Ultimately, safety is the arbiter of reliability for nuclear power, since a major plant accident would seriously jeopardize public acceptance of the technology. The early designers of commercial nuclear systems could not have perceived that "nuclear" would come under attack and raise doubts in the public mind to the extent that it has. Fortunately the nuclear industry undertook responsibility for public protection to a degree not known in any other comparable industry, and the standards which were set have been developed and tightened ever since. A number of guides and rules for good nuclear practice have been produced. Undoubtedly the nuclear industry is the most regulated industry in history, and this costs money. Millions of dollars are spent on research and the design, construction and operation of safe systems. In any nuclear plant you will find an army of planning and maintenance employees. But set this against the cost of outage and you get a better perspective. Earlier the figure of $140,000 was quoted as the cost for a day's outage on one Pickering G.S. unit. This represents $51 million per year and is only the difference in the fuel cost of coal and uranium. For the utility and the public the investment in safety has paid off in availability and hence the costs of our electricity.

This paper has no more than scratched the surface of the regulation/availability relationship, and it has avoided any comparison of nuclear and fossil-fuel generating station practices. However, the application of the safety principles and practices that have been developed, to thermal-electric stations generally are worthy of serious consideration on a cost-incurred to benefit-gained basis. One additional point must be recognized. We are now into the age of environmentalism. New standards are going to be imposed, not just for new technology such as nuclear power, but also for other technologies. The designers and operators of fossil stations could do worse than take a look at nuclear standards, simply as insurance for the future.

REFERENCE

NOTE: Outages of less than three days are not indicated.

**Pickering Nuclear Generating Station Power History Units 1-4**

**Unit One**
- Net capacity factor - 1975: 80.2%
- In-service date: 29 July 1971
- Number of sudden outages in 1975: 2

**Unit Two**
- Net capacity factor - 1975: 58.0%
- In-service date: 30 December 1971
- Number of sudden outages in 1975: 3

**Unit Three**
- Net capacity factor - 1975: 57.5%
- In-service date: 1 June 1972
- Number of sudden outages in 1975: 3

**Unit Four**
- Net capacity factor - 1975: 23.8%
- In-service date: 17 June 1973
- Number of sudden outages in 1975: 2

Maximum continuous rating gross = 540 MWe
Maximum continuous rating net = 514 MWe