LIC 1/20/81



UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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| In the Matter of | ) | |
| | ) | |
| METROPOLITAN EDISON COMPANY | ) | Docket No. 50-289 |
| | ) | (Restart) |
| (Three Mile Island Nuclear Station, Unit No. 1) | ) | |
| | ) | |

LICENSEE'S TESTIMONY OF

SALOMON LEVY

IN RESPONSE TO UCS CONTENTION NO. 13

(ALTERNATIVE ACCIDENT SEQUENCES/CLASS 9 ACCIDENTS)

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## OUTLINE

The purposes and objectives of this testimony are to respond to UCS Contention 13, which asserts that the design of TMI-1 does not provide protection against so-called "Class 9" accidents, that the process of selecting events to be considered is faulty, and that there is therefore no reasonable assurance that TMI-1 can be operated without endangering the health and safety of the public. The testimony demonstrates, in a semi-quantitative way, that the design, safety evaluation and licensing process employed for TMI-1 provides capability to cope with events not analyzed and with events beyond design basis accidents. Accident sequences having a reasonable nexus to the TMI-2 accident are identified, and the prevention and mitigation of the consequences of such accidents are discussed qualitatively. A quantitative, though approximate, evaluation is presented of design modifications, improvements in operator training, and other actions taken in response to the TMI-2 accident, to illustrate the substantial increase in the margin of safety achieved for sequences having a nexus to the TMI-2 accident.

# INDEX

INTRODUCTION

This testimony, by Salomon Levy, is addressed to the following contention:

UCS CONTENTION NO. 13

> "The design of TMI does not provide protection against so-called "Class-9" accidents. There is no basis for concluding that such accidents are not credible. Indeed, the staff has conceded that the accident at Unit 2 falls within that classification. Of the realm of possible accidents, the staff's method of determining which fall within the design basis accidents and those for which no protection is required is faulty in that the design basis accidents for TMI do not bound the credible accidents which can occur. Therefore, there is not reasonable assurance that TMI-1 can be operated without endangering the health and safety of the public and resumption of operation should not be permitted."

RESPONSE TO UCS CONTENTION NO. 13

BY WITNESS LEVY:

The design, safety evaluation and licensing process of TMI-1 requires the evaluation of a very large number of specific events. The events considered cover a broad range of frequency of occurrence, i.e., from normal operation or high frequency conditions to design basis accidents or very low frequency conditions. Events to be considered are not selected through application of a specific method or a numeric probability goal which subdivides events into two categories -- credible and incredible; instead, the selection process is based upon years of evolving experience, and the careful composite engineering judgment of the NRC, ACRS, and the nuclear industry.* The events examined for TMI-1 are discussed in Sections 14 and 3.2.3 of the TMI-1 Final Safety Analysis Report (FSAR)(1) and in Section 8 of the TMI-1 Restart Report (2).

In addition to identifying the specific events to be considered, the design, safety evaluation and licensing process

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* In recent years, a listing of events to be considered has been generated and is included in Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (Revision 2), issued in September 1975. In some instances, specific plant designs or site features (for example, aircraft impact) require consideration of additional design basis accidents. Accidents in this listing agree closely with the accidents recommended for evaluation in the American Nuclear Society Pressurized Water Reactor Criteria (ANSI No. 18.2).

also prescribes different levels of fuel integrity to be maintained or releases of radioactivity not to be exceeded over the entire spectrum of analyzed sequences. The range of occurrences examined for TMI-1 can be subdivided into four major categories:

- normal conditions, which occur at a very high to high level of frequency and for which a conservative margin is provided towards avoiding fuel failure or release of radioactivity.

- anticipated operational events, which occur at a high to moderate frequency (<u>i.e.</u>, once to several times per year) and which are required to not produce fuel failure or release of radioactivity.

- abnormal operational events with a moderate to low probability of occurrence (in the range of once in ten to one hundred years), which are required to not perforate more than one percent of the fuel rods and to release only the gaseous radioactivity present in the gap between fuel cladding and fuel pellet of such failed rods.

- design basis accidents with a very low probability of occurrence (in the range of once per one thousand to ten thousand years), for which fuel cladding peak temperature or fuel peak specific energy content must be shown to fall below prescribed values to again limit the number and type of fuel failures and the releases of radioactivity that might ensue.

All the above postulated events were evaluated for TMI-1 using conservative assumptions. "Conservative" means that the performance of the TMI-1 plant and safety systems are deliberately underestimated and that the fuel conditions and radioactive consequences are deliberately overestimated. One example of such conservatism is the assumption of the worst or most limiting single failure in any safety-related system or function utilized in the required analyses. Another example is the assumption that whenever Departure from Nucleate Boiling (DNB) occurs, the fuel cladding fails. In fact, DNB corresponds to the point at which the fuel rod surface begins to be locally blanketed with steam; the cladding temperature at that location rises due to the steam blanketing but does not exceed the 3300°-3500°F temperature threshold at which cladding perforation may occur. (3) This will be discussed further later. One final illustration of the "conservative" approach taken in licensing evaluations is the finding that licensing calculation methods overpredict by about 500° to 1000° F the temperatures measured at the Loss-of-Fluid Test (LOFT) facility during simulated loss-of-coolant accidents. The impact of such conservative assumptions is that the predicted consequences are overestimated substantially or that the probability of the predicted consequences actually occurring is high by at least one order of magnitude per year.

An effort was made herein to approximately quantify the design, safety evaluation and licensing process employed at TMI-1. Figure 1 and Table 1 show the four major groups of occurrences examined for TMI-1 and the fuel failures estimated for such events in the TMI-1 FSAR. The first group, normal conditions, includes the various modes of possible plant operation, e.g., refueling, shutdown, partial power and full power. During normal operation, fuel failure is conservatively estimated to occur when the margin to DNB goes below 1.3. (The factor 1.3 is used instead of 1.0 to provide a very high level of confidence that a very large percentage of the fuel rods are in no jeopardy of experiencing DNB.) At 100 percent power at TMI-1, the DNB margin is expected to be above 2.0 for the most probable conditions. At reduced power and during shutdown and refueling, the margin would be well above 2.0. For the worst nuclear, thermal and mechanical conditions during normal TMI-1 operations, the DNB margin is expected to exceed 1.55, even at the maximum assumed overpower limit of 114 percent(2). The most probable margin to fuel failure and radioactive release in this very high to high frequency range of events is thus estimated to be 1.5 (or 2.0 divided by 1.3) and to never fall below 1.2 (or 1.55 divided by 1.3). These margins are indicated by the cross hatching in Figure 1 in the very high to high frequency range of occurrence.

The second group of events, which range from high to moderate frequency, includes many anticipated transients, as listed in Table 1. The transients are produced by system and control disturbances or component malfunctions. Because the DNB margin for all such events is required to be above 1.3, no fuel failure or radioactive release is expected. Table 1 and Figure 1 illustrate the results of the evaluations of such anticipated transients.

The third group of events, which occur from moderate to low frequency, result from a failure of components or operator error with and without an anticipated transient, or from a small break. This category of events includes sequences having reasonable nexus to the TMI-2 accident since the TMI-2 accident resulted from the anticipated transient of loss of main feedwater combined with failure of a relief valve to reclose. The events considered in this group are listed in Table 1, with their impact on fuel integrity. The expected release of fission products is cross hatched in Figure 1. In this category of events, the number of fuel failures is expected (and required) to be small.

The fourth group of events occur at low to very low frequency. It includes loss-of-coolant accidents (LOCAs) beyond small breaks and extending over the entire spectrum up to and including rupture of the largest pipe diameter in the various plant systems; control rod assembly drop or ejection and fuel handling accidents are also considered. For all such events, fuel failures are required to be limited and activity released must be kept well below the values specified in Regulatory Guide 1.4. For example, in the case of loss-of-coolant accidents, this is achieved by requiring Emergency Core Cooling System (ECCS) performance in accordance with 10 CFR 50.46. Even so, under Regulatory Guide 1.4, issued in June 1974, twenty-five percent of the equilibrium iodine and one hundred percent of the equilibrium radioactive noble gas inventory developed from maximum full power operation is required to be assumed to be immediately available for release from the reactor containment for the purpose of calculating offsite dose consequences. The estimated fuel failures and radioactivity releases for this final group of events are indicated in Figure 1; as discussed above, a conservative demonstration that fuel failures (thus, source terms) are considerably below the values of Regulatory Guide 1.4 is required.

As illustrated in Figure 1, the design, safety evaluation and licensing process employed at TMI-1 requires the consideration of a very large number of specific events and the assurance that fuel failures and radioactive releases are kept within specified limits which are more and more stringent as the frequency of occurrence of the event increases. It was not intended (nor is it possible) to consider all events that could conceivably occur in a nuclear power plant. The intent was to consider enough event types over a large and defined spectrum of frequency to envelope the consequences of events not considered over the range of frequency shown in Figure 1. Thus, the process does not stop at this point; rather, the process ensures additional capability to cope with events not analyzed and with events which might occur at lower frequencies than postulated. The source of this additional capability is the defense-in-depth approach employed in all light water power reactors.

The defense-in-depth concept relies upon multiple barriers to prevent radioactive fission product releases. The defense-in-depth approach is supported by diverse but interrelated programs for redundancy in protection and safety systems, industrial standards and regulatory guides, quality assurance programs, surveillance and preventive maintenance requirements, and operator training.

The first barrier of defense-in-depth is the ceramic form of fuel and the fuel cladding employed. The ceramic pellets of uranium dioxide retain about 98% of the radioactivity generated by the nuclear fission process. Only approximately 2 percent of the radioactivity in gaseous form is present in the gap between fuel cladding and pellet. It is released upon failure of fuel cladding. Even when the fuel cladding is heated to a temperature several hundred degrees above the normal operating temperature, only the radioactivity in the gap space plus a small portion of the volatile fission products located at the surface of the ceramic pellets are available for release. Even under the most extreme case of fuel melt, still only 15 to 25 percent of the total radioactivity is released from the molten portion of the fuel.

Any radioactivity released from the fuel is confired by the reactor coolant system piping and vessel, as long as they remain intact. This second barrier is designed and constructed to the highest quality standards. Finally, even if the reactor coolant system boundary is breached, the containment building serves as a third barrier. The containment houses the reactor coolant system and is designed to essentially confine any radioactivity that should escape from the reactor coolant system.*

The multibarrier concept of defense-in-depth ensures reserve capability. For example, focusing upon events of moderate to low frequency (i.e., sequences having reasonable nexus to the TMI-2 accident), there are still one to two barriers at the termination of the event to minimize the transport of fission products outside the plant boundary, as indicated at the top of Figure 1. In the range of moderate to

* Both siting and onsite and offsite emergency planning provide additional barriers and further defense-in-depth. These additional barriers are particularly important in consideration of beyond design basis events.

low frequency of occurrence, only a few fuel cladding failures are permitted. Thus, for those events involving a small break or leading to a small break of the reactor coolant system boundary:

- the integrity of the containment is preserved and the containment is still available to prevent significant radioactive release outside the plant.

- the pressure in the containment for the worst case analyzed will be below its design pressure. (Even if the containment were to reach design pressure, it has a reserve capability and will therefore withstand static pressures at least twice the design pressure before it might fail.)

- the fission product releases would be quite minimal compared to those prescribed by Regulatory Guide 1.4 and for which the containment has been designed.

Recently, the possibility of even more capability in terms of fission products released within the containment has been highlighted in the August 14, 1980 letter from W. R. Stratton, A. P. Malinauskas, and D. O. Campbell to Chairman J. Ahearne of the NRC (4). The letter points out that the radioactive iodine released to the environment during the TMI-2 accident was about 100,000 to 1,000,000 times less than the corresponding release of Xenon-133, and recommends that "the frequently quoted fission products escape assumptions (from TID-14844 in 1962 to the more recent Regulatory Guides 1.3 and 1.4, and the Reactor Safety Study, WASH-1400) should be reexamined. The present assumptions grossly overstate iodine release from a reactor site in many types of loss-of-coolant accidents....." This topic was addressed further at a November 18, 1980 Commission briefing on Iodine Release from Accidents and Estimates of Consequences of Nuclear Accidents (5). At that briefing, C. Starr, M. Levenson, and I. Wall of the Electric Power Research Institute (EPRI) discussed realistic estimates of the consequences of nuclear accidents. Their comments, which have been amplified in a recent publication by M. Levenson and F. Rahn (6), reveal that the radiation releases from past controlled experiments and reactor accidents are one to two orders of magnitude less than the escape assumptions employed in nuclear safety and risk evaluations. This difference is attributed to inappropriate recognition of "a number of physical processes which are always operative and can be counted on to limit the consequences of a reactor accident." In particular, the researchers emphasize that the chemical reaction and solubility of volatile fission products, particularly iodine, in water are not accounted for properly, and that aerosol coalescence and agglomeration, steam condensation, and deposition on surfaces will act "to reduce the magnitude of the fission product release and change the character of the release in that iodine and particulates are greatly reduced relative to the noble gases."

The preceding discussion has particularly emphasized the extra capability provided by the containment for events having reasonable nexus to the TMI-2 accident. However, as shown in Figure 1 and Table 1, similar additional capability can be demonstrated for other events of moderate to low frequency of occurrence which do not lead to a small break of the primary cooling system. For example, in cases such as uncontrolled control rod group withdrawal, the primary coolant system barrier would remain intact and provide equivalent or extra capability to cope with any fission release during an uncontrolled control rod group withdrawal.

The multibarrier concept also ensures reserve capability for events with a low to very low frequency of occurrence, as illustrated in Figure 1. The capability to cope does not drop to zero for accidents beyond the design basis events included in Figure 1 and Table 1, because the design process requires that the integrity of one barrier still be available for protection after the occurrence of a design basis accident. Studies such as WASH-1400 (7) and the German risk assessment (8) have examined accident sequences beyond design basis events; they indicate that light water nuclear power plants can cope with events beyond design basis accidents. In other words, even though some accidents are not required to be analyzed or considered, it is not true that no protection is available or required for them. The present process, which identifies specific events to be considered and imposes stringent limits and conditions on their consequences, ensures additional capability for other accidents.*

In the case of TMI-2, an abnormal transient in the moderate to low frequency range degenerated to the point where it resulted in more fuel damage than one would have expected from Figure 1 and Table 1. One of the issues which the NRC Staff and the industry have addressed since the TMI-2 accident is how to ensure that, in the future, a TMI-2 type event will have a much lower frequency of occurrence and/or produce fuel failures limited within the range shown in Figure 1 and prescribed by the present process.

The present process would put foremost emphasis on limiting the number of fuel failures to a very small number. This is the correct approach because, at the probability level of sequences having reasonable nexus to the TMI-2 accident, it is clearly preferable to prevent the occurrence of a large number of fuel failures rather than attempt to cope with the degraded conditions of the TMI-2 accident.

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* An alternate process would be to consider events beyond design basis accidents and to impose less stringent limits on their consequences. This has not been done to date because the methods of performing such evaluations have not been fully developed or approved for use in the licensing process.

Several actions have been taken since the TMI-2 accident to prevent its reoccurrence. This testimony is not intended to comprehensively address all such actions. However, in the section which follows, an effort is made to quantify the increased protection resulting from such actions, rather than speaking in qualitative terms only. Many of the numbers that follow are judgmental. However, even if one disagrees with the absolute values proposed, there is no doubt that substantial strides have been made to prevent the reoccurrence of the TMI-2 sequence and the occurrence of other sequences having a nexus to TMI-2 accident.

In the original TMI designs, a loss of main feedwater usually led to the actuation of a relief valve (PORV); the probability of such a relief valve failing to reclose, and thus cause a LOCA, is estimated to be about once every fifty times for this event sequence. (7) Changes made to the main and emergency feedwater systems and differences between the design of the TMI-1 polishing demineralizers system and that of TMI-2 have reduced the probability of the transient being initiated. Also, the pressure set point of the relief valve and of the reactor protection system will be changed and anticipatory reactor trips will be installed so that the relief valve will not be expected to open for a loss of main feedwater transient. All these changes are judged to reduce the frequency of the initiating event which results in the opening of the PORV by a factor of at least 2 to 3.

In the revised design, an inadvertent opening of the relief valve and its failure to reclose or a small independent reactor coolant system break would be required to reproduce the TMI-2 accident sequence. It is estimated that a relief valve will inadvertently open at a frequency of once per 1000 per year. (7) Combining this figure with the once in fifty times for the relief valve to stick open, the probability of a small break being produced by a relief valve following a loss of main feedwater transient is now estimated to be about once per 50,000 times instead of once per 50 times. However, the probability of a small break occurring independently of relief valve actuation is estimated at about once per 1000 reactor years (7), so that the net effect of resetting the relief and reactor protection pressures to avoid a relief valve opening has, in fact, reduced the probability of the TMI-2 accident and its analogs by a factor of about 20 (small break probability divided by probability of relief valve to reclose).

The planned installation of a saturation meter and the associated procedures and instrumentation implemented to help the operator recognize inadequate core cooling (9) will improve the emergency cooling injection availability by a factor of 2 to 3. The increased operator training required since the TMI-2 accident and planned staffing augmentation (including additional licensed personnel and a Shift Technical Advisor) will provide for another improvement factor of 3 to 5 (equivalent to that available for small breaks produced by other means than relief valves). Thus, I conclude that all the changes described above provide for an overall improvement factor of about two to three orders of magnitude for a TMI-2 type accident initiated by loss of main feedwater.

The probability of a stuck open relief valve for other transients involving the opening of the relief valve is the product of the probability of the relief valve being demanded to open times the probability of the valve failing open on demand. The raising of the relief valve setpoint from 2255 to 2450 psig, the lowering of the high pressure reactor trip setpoint from 2390 to 2300 psig, and the installation of an anticipatory reactor trip on turbine trip and/or loss of all main feedwater at TMI-1 has considerably reduced the number of demands on the relief valve. A preliminary estimate indicates that the probability of the relief valve opening on an overpressure transient, or due to operator action, or due to instrumentation control faults is about once per 50 reactor years of operation. When this figure is multiplied by the probability of once per 50 times of the relief valve sticking open on demand, the probability of a stuck open relief valve from all causes is calculated to be about once per 2500 reactor years of operation; this compares favorably to the probability of once per 1000 reactor years of operation for occurrence of a loss-of-coolant accident of that size from a leak in the reactor primary system (7).

The preceding probabilistic estimates are not intended to address all actions taken since the TMI-2 accident or all event sequences having a nexus to the TMI-2 accident. Substantial testimony is being presented in this proceeding by the witnesses of Licensee and the NRC Staff on the many actions taken in response to the TMI-2 accident. The estimates presented here were generated to provide a quantitative illustration of the increase in the margin of safety achieved in many typical sequences having a nexus to the TMI-2 accident.

In summary, the present design, safety evaluation and licensing process employed at TMI-1 requires the consideration of a very large number of specific events and the assurance that fuel failures and radioactive releases are kept within specified limits which are more and more stringent as the frequency of occurrence of the event increases. All the events are evaluated using "conservative" assumptions. The process also relies upon a defense-in-depth approach which provides additional reserve capability to cope with events not analyzed and with events which might occur at lower frequencies than postulated. This defense-in-depth approach, together with considerably overestimated releases of fission products, accounts for the "negligible effect on the physical health of individuals" attributable to the TMI-2 accident, reported by the President's Commission on the Accident at Three Mile Island (10) and the NRC Special Inquiry Group (11). The sequences of events having a reasonable nexus to the TMI-2 accident have a moderate to low probability of occurrence. The correct approach at that probability level is to prevent the occurrence of a degraded core rather than to cope with such a degraded core. A substantial number of actions have been taken since the TMI-2 accident. Some of the actions taken provide for an overall improvement factor of about two to three orders of magnitude for a TMI-2 type accident initiated by loss of main feedwater. Similarly, substantial improvement has been made for other overpressure transients involving relief valves; overall, substantial actions have been taken to ensure that sequences of events having a reasonable nexus to the TMI-2 accident are terminated long before the core reaches a degraded condition.

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(4) Letter by W. R. Stratton, A. P. Malinauskas, D. O. Campbell to J. Ahearne, dated August 14, 1980.

(5) Nuclear Regulatory Commission Meeting on Iodine Release from Accidents and Estimates of Consequences of Nuclear Accidents, November 18, 1980, Washington, D.C.

(6) The Need for Realistic Estimates of the Consequences of Nuclear Accidents, by M. Levenson and F. Rahn, EPRI, December 1980.

(7) Reactor Safety Study, WASH-1400 (NUREG-75/014), October 1975.

(8) Deutsche Risikostudie, Verleg TUV Rheinland, 1980.

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(10) Report of the President's Commission on the Accident at Three Mile Island, October 1979.

(11) Three Mile Island - A Report to the Commissioners and the Public, Vol. I, Nuclear Regulatory Commission Special Inquiry Group, January 1980.

## TABLE I. EVENTS CONSIDERED AT TMI-1

### I. NORMAL OPERATION (VERY HIGH TO HIGH FREQUENCY OF OCCURRENCE)

| | |
|---|---|
| Shutdown, Refueling, Startup, Partial, Full Power Conditions and Normal Load Following | Margin to fuel failure always in excess of 1.2 and most probably 1.5 |

### II. ANTICIPATED TRANSIENTS (HIGH TO MODERATE FREQUENCY OF OCCURRENCE)

| Event Considered | Impact on Fuel |
|---|---|
| Decrease in feedwater temperature | No fuel failure |
| Increase in feedwater flow | No fuel failure |
| Steam regulatory malfunction | No fuel failure |
| Loss of external electric load and/or turbine load without runback | No fuel failure |
| Closure of steam isolation valves | No fuel failure |
| Loss of condenser vacuum | No fuel failure |
| Loss of main feedwater flow | No fuel failure |
| Chemical addition malfunction | No fuel failure |
| Inadvertent operation of ECCS | No fuel failure |

### III. ABNORMAL TRANSIENTS (MODERATE TO LOW FREQUENCY OF OCCURRENCE)

| Event Considered | Impact on Fuel |
|---|---|
| Rod withdrawal accident at rated power operation | No fuel failure |
| Loss of external electric load and/ or turbine load without runback | No fuel failure |
| Loss of non-emergency AC power to the station auxiliaries | No fuel failure |
| Loss of coolant flow | No fuel failure |
| Startup accident | No fuel failure |

## IV. ACCIDENTS (LOW TO VERY LOW FREQUENCY OF OCCURRENCE)

| | |
|---|---|
| Feedwater piping break (entire spectrum) | No fuel failure |
| Reactor coolant pump shaft break or seizure | 1% fuel failures and associated gap activity release |
| Stuck-Out, Stuck-In, or Dropped-In Control Rod | No fuel failure |
| Fuel handling accident | Failure of 56 fuel rods and associated gap activity release |
| Steam line break (entire spectrum) | 1% fuel failures and associated gap activity release |
| Control rod ejection | 17.5% fuel failure and associated gap activity release |
| Primary system piping break (entire spectrum) | 100% fuel failure and associated gap activity release assumed |

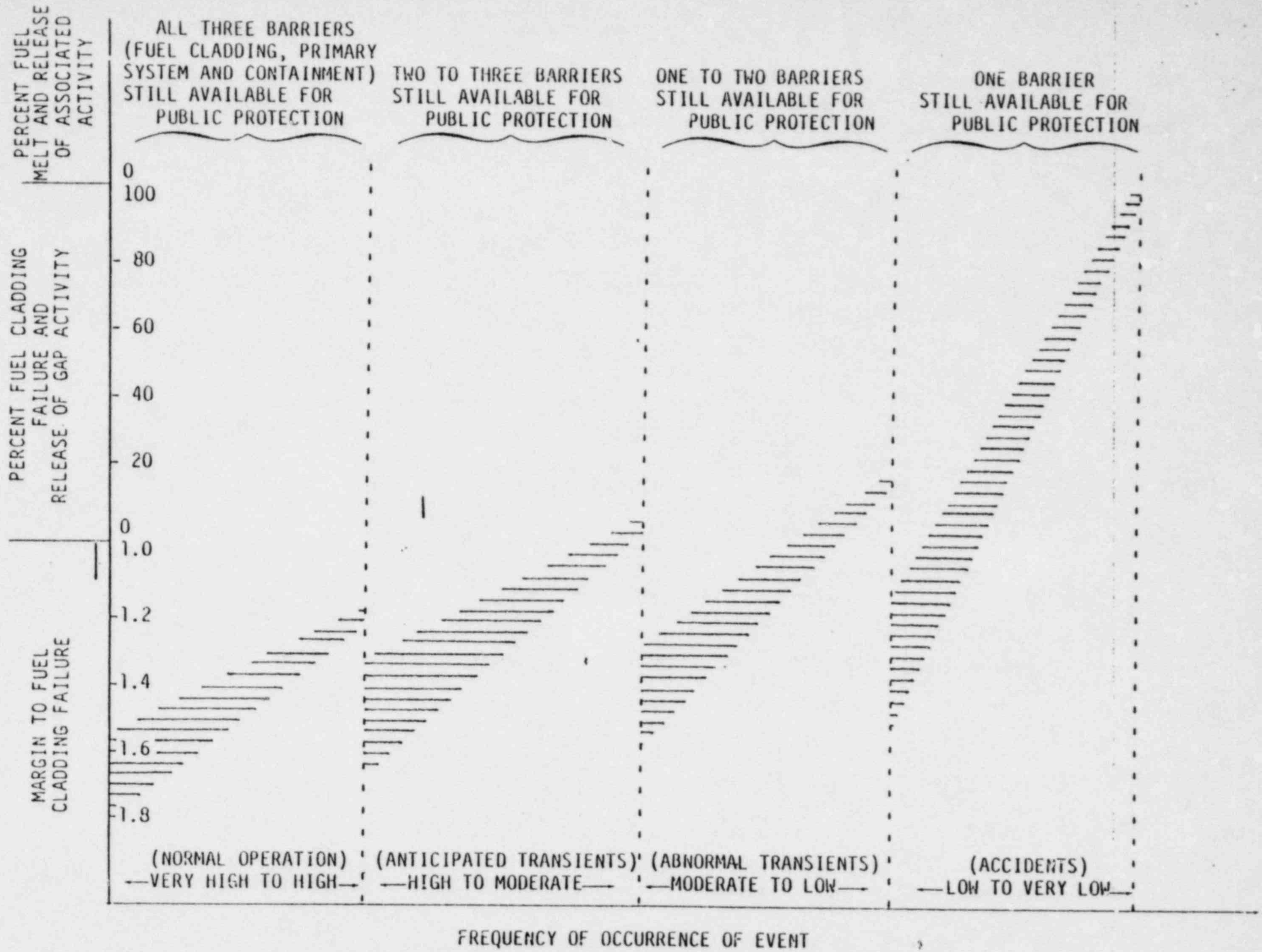


FIGURE 1. REPRESENTATION OF DESIGN AND LICENSING PROCESS

SALOMON LEVY

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| Business Address: | S. Levy Incorporated<br>1901 South Bascom Avenue, Suite 725<br>Campbell, California 95008 |
| Education: | B. S., Mechanical Engineering, University of California at Berkeley, 1949.<br>M.S., Mechanical Engineering, University of California at Berkeley, 1951.<br>Ph.D., Mechanical Engineering, University of California at Berkeley, 1953. |
| Experience | President, S. Levy Incorporated, September 1977 to present. Independent engineering consultant to several power utilities, national laboratories, Electric Power Research Institute, Nuclear Regulatory Commission Research Division and several power equipment manufacturers; consultant to the staff of the President's Commission on the Accident at Three Mile Island and member of the Industry Advisory Board to Three Mile Island-2 recovery operations.<br><br>General Manager, Boiling Water Reactor Operations, General Electric Company, April 1975 to September 1977. In this position, was responsible for all the engineering and manufacturing aspects of General Electric's nuclear power business.<br><br>General Manager, Boiling Water Reactor Systems Department, General Electric Co., 1973 to 1975. Responsible for the design and development of nuclear systems including fuel and for the manufacturing of control and instrumentation systems.<br><br>General Manager, Nuclear Fuel Department, General Electric Co., 1971 to 1973. Responsible for the design, development and manufacture of nuclear fuels for light water reactor systems and for reprocessing of irradiated fuel. |

Manager and Design Engineer, Atomic Power Equipment Department, General Electric Co., 1968 to 1971. Responsible for the design engineering of all nuclear systems being offered by General Electric and for all project management functions associated with domestic nuclear systems.

Manager, Systems Engineering, Atomic Power Equipment Department, General Electric Co., 1966 to 1968. Responsible for conceiving and defining nuclear power plant systems for all requisition and proposal plants and all near-term improvements in nuclear power plant systems.

Manager, Heat Transfer and Reactor Projects, Atomic Power Equipment Department, General Electric Co., 1959 to 1966. Responsible for heat transfer and fluid flow development in boiling water reactors.

Advance Nuclear Specialist, Atomic Power Equipment Department, General Electric Co., 1956 to 1959. Involved with advanced reactor concepts such as fast oxide breeders.

Supervisor, Atomic Power Equipment Department, General Electric Co., 1954 to 1956. Responsible for the design, safeguard analysis and development of small atomic power plants and test reactors.

Engineer Analyst, Knolls Atomic Power Laboratory, 1953 to 1954. Worked with steam boilers and superheaters.

Research Engineer, University of California at Berkeley, 1950 to 1953. Conducted boundary layer studies of heat transfer in high speed flight.

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| Honors and Professional Affiliations: | Member, National Academy of Engineering. |
| | Fellow, American Society of Mechanical Engineers; Past Chairman, ASME Heat Transfer Division. |
| | ASME Heat Transfer Memorial Award (1966). |
| | ASME/AIChE Heat Transfer Conference Award (1963). |
| | Member, Editorial Board, Nuclear Science and Engineering. |
| | Past Vice-Chairman, Management Committee, CBI Nuclear, Memphis, Tenn. |
| | Past Member, Editorial Board, International Journal of Heat and Mass Transfer, Heat Transfer in Russia, Heat Transfer in Japan. |
| | Past Member, AEC Task Force on Emergency Core Cooling. |
| | U.S. Delegate, 1964 International Conference on Peaceful Uses of the Atom. |
| Publications: | Representative publications include: |
| | "Heat Transfer to Constant-Property Laminar Boundary-Layer Flows with Power-Function Free-Stream Velocity and Wall-Temperature Free-Stream Velocity and Wall-Temperature Variation," Journal of Aeronautical Sciences. 19, No. 5, 341-348 (May 1952). |
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January 20, 1981



UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

| | | |
|---|---|---|
| In the Matter of | ) | |
| | ) | |
| METROPOLITAN EDISON COMPANY | ) | Docket No. 50-289 |
| | ) | (Restart) |
| (Three Mile Island Nuclear | ) | |
| Station, Unit No. 1) | ) | |

## CERTIFICATE OF SERVICE

I hereby certify that copies of "Licensee's Testimony Of Salomon Levy In Response To UCS Contention No. 13 (Alternative Accident Sequences/Class 9 Accidents)" were served upon those persons on the attached Service List by deposit in the United States mail, postage prepaid, this 20th day of January, 1981.

Delissa A. Ridgway

Delissa A. Ridgway

Dated: January 20, 1981

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