UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket Nos. 50-247-SP
CONSOLIDATED EDISON	)	50-286-SP
OF NEW YORK (Indian Point, Unit 2)	)	
	)	
POWER AUTHORITY OF THE STATE	)	
OF NEW YORK (Indian Point, Unit 3)	)	March 22, 1983

TESTIMONY OF ROBERT M. BERNERO
ON SEVERE ACCIDENT SOURCE TERMS

Q.1 Please state your name and position with the NRC.

A.1 My name is Robert M. Bernero. I am the Director of the Accident Source Term Program Office in the Office of Nuclear Regulatory Research at the U.S. Nuclear Regulatory Commission.

Q.2 What are your responsibilities in that position?

A.2 My responsibilities in this position are to assure that Source Term related research results are implemented in policy and regulatory practice in a timely manner.

Q.3 Have you prepared a statement of your professional qualifications?

A.3 Yes, the statement of my professional qualifications is attached to this testimony.

Q.4 What is the purpose of your testimony?

A.4 This testimony provides the Staff's assessment of the status of current research in the subject matter raised in the testimony of Drs. Stratton and Rodger, and Mr. Potter, bound into the record following T.R. 8169, i.e., the quantity and characteristics of the radionuclides released following an accident, often referred to as the radiological "source term." (hereinafter referred to as the Stratton testimony)

Q.5 How is an assessment of the Source Term conducted?

A.5 The words "source term" are often used as a simple term suggesting that severe (severe core damage or core melt) accidents in nuclear reactors can be characterized by a single source term; and one might speak of reducing that source term by some factor, say of 10 or 5 or 100. In fact, each severe accident in a nuclear reactor of a specific design has a characteristic scenario and, therefore, has a characteristic pathway by which the radioactive material is moved from the core through the reactor coolant system out into the containment or other buildings and ultimately out into the biosphere. Each severe accident scenario then has a characteristic source term describing the amount and form of the radioactive materials which are released to the environment. In order to assess radiological source terms comprehensively, one must examine each of these accident scenarios and study the behavior of each of the principal species of radionuclides in each phase of these scenarios. Thus, a rather complex assessment is made, there are many source terms, and many complicated potential changes in source terms.

Q.6 Please provide the Staff's assessment of the Source Term questions and research activities discussed in the Stratton Testimony.

A.6 The Stratton testimony draws attention to the intense research program currently underway which is intended to resolve some of the questions concerning accident source terms identified in part by Dr. Stratton. We agree with the Stratton testimony's assessment that the accident source term methodology employed in the Reactor Safety Study (WASH-1400) results in conservative predictions, i.e., in overestimates of the quantity of fission products released to the environment. To the extent that the RSS methodology was employed, therefore, such conservatisms are incorporated in the Indian Point Probabilistic Safety Study (IPPSS) and in the staff's consequence estimates as embodied in the testimony of staff witnesses Meyer and Acharya. This agreement with Stratton et al. is based on the existing knowledge that several physical processes which would affect some degree of retention of radionuclides in various parts of the damaged facility were neglected in the RSS methodology for assessing radionuclide release and transport. The Stratton testimony draws heavily on quotations from NRC-sponsored research reports to establish this fact, so that repetition herein is not necessary. However, at the present time, we cannot agree with the next step in the Stratton testimony, i.e., the subjective estimates of quantitative "reduction factors" asserted on the basis of a qualitative description of a partial list of the physical processes affecting radionuclide release and transport. These estimates of reduction factors to be applied to the IPPSS are based solely on the subjective judgment of the authors of the Stratton testimony, and do not necessarily follow from the common ground of the general technical consensus described above. Although the current research effort may provide substantiation for some of the assertions of the Stratton testimony, the conclusions in the Stratton testimony regarding revised source terms are premature at this time. The staff does believe that the technical data available today strongly suggest that the WASH-1400 models for radionuclide behavior in severe accidents overestimate the amount of such material that would be released to the environment in such an accident. As I will explain later, the staff and its contractors are engaged in a substantial research program to obtain better data and models for estimating these source terms. It is our involvement in this work and the complexity of the analyses at this time that prompt us to say that selecting and using a new source term model at this time is premature.

The following examples demonstrate the reasons for the staff's inability to endorse the conclusions of the Stratton testimony.

1. The TMI Experience

A major factor influencing the conclusions in the Stratton testimony is the interpretation given to the TMI accident. The Stratton testimony (p. 15ff) sets up a comparison of the TMI event with release categories PWR-1 and -2, and in Table 4 (p. 53) with a WASH-1400 "average" and "lowest" release. On p. 17 the Stratton testimony notes agreement between the releases from the core for PWR-1 and -2 and the TMI observation, and then notes the lack of agreement of the releases from the containment. However, it should be noted that the RSS calculations are for accidents involving containment failure, while no failure of containment occurred at TMI. The PWR-1 and -2 release categories of the RSS are dominated by accident sequences involving catastrophic failure of containment within 30 minutes of the initiation of the accident. In the TMI-2 accident, the principal releases were through the letdown and makeup system and the auxiliary building vent header many hours, even days after the accident. A direct comparison of the atmospheric releases from postulated events such as PWR-1 and PWR-2 release accidents with the TMI-2 accident where the atmospheric leakage from the containment was negligible cannot provide much insight concerning the effects of fission product transport behavior. This is not to say that valuable insight cannot be obtained from the TMI experience. Careful examination of the TMI data is expected to provide a useful reference for some mechanistic fission product release and transport models. However, some basic work, i.e., the examination of the TMI core, and determination of its fission product inventory and distribution is still to be completed. Completion of the TMI core examination is perhaps 2 years away.

2. Cesium Iodide

While the Stratton testimony almost quotes (Stratton, p. 22) the staff's conclusion that cesium iodide is the most likely predominant form of iodine released from the reactor core (NUREG-0772), Stratton concludes, without further quantification of the physical/chemical processes involved, that this change from the RSS assumption permits reduction of the quantity of iodine released to the environs. A factor apparently weighing heavily in this assessment is the repeated emphasis on the solubility of CsI and other "metallic iodides" (Stratton, p. 19) as opposed to a presumed "insolubility" of I_2 ascribed to the RSS model. The authors of the Stratton testimony apparently interpreted the RSS methodology to ascribe "near-noble gas" behavior to I_2 (Stratton, p. 21). In reality, the RSS methodology ascribes very high solubility to I_2. For an accident sequence of the TMI type (i.e., the release from the core contacts water prior to release to the environment, and pH adjusted containment sprays are operational) the RSS's CORRAL code calculates that the iodine concentration in the liquid phase is more than 10,000 times higher than that in the containment atmosphere (WASH-1400, App. VII). Therefore, it is necessary to perform a careful examination of the dynamic behavior of radionuclide transport mechanisms based on the expected thermal hydraulic conditions in the primary system and containment in order to assess the effects of the chemical form of the iodine.

3. Retention in the Primary System

The Stratton testimony correctly points out that the RSS analyses of fission product transport do not account for agglomeration and deposition of fission products in aerosol form within the primary system. We agree that this is a known conservatism of the RSS and other analyses employing the RSS methodology. The Stratton testimony discusses the potentially important phenomena in the reactor vessel (Stratton, p. 31ff). However, the Stratton testimony quotes from a 1977 publication which describes the processes important in a "terminated LOCA," i.e., a design basis accident for which no substantial fuel degradation or melting is assumed (i.e., the TRAP-LOCA code). We believe that a more appropriate basis for estimates of primary system retention is the next level of development of the TRAP model, i.e., the TRAP-MELT code which addresses core melt conditions (NUREG/CR-0632, 1979). Calculations of primary system retention during core-melt accidents using the latest version of this code are currently in progress. Although it is premature to make generic conclusions on the basis of these calculations, the results (as stated in NUREG-0772) indicate that the degree of radionuclide retention in the primary system is highly dependent on the specific accident sequence and the specific reactor design, and suggest that categoric "reduction factors" cannot be supported at the present time.

4. Fission Product Behavior in Containment

In order to characterize the behavior of fission products in the containment, the Stratton testimony refers back to the Containment System Experiment (CSE) of the late 1960s (Stratton, p. 40ff) to demonstrate the effectiveness of containment spray in removal of elemental iodine and aerosol particles. The authors of the Stratton testimony reject the quantitative predictions of radionuclide behavior in the containment provided by the RSS's CORRAL code. However, it should be noted that the CORRAL code is based on an empirical fit to the CSE data. Better predictions of radionuclide behavior in containment are now being validated, such as the NAUA-4 code; but the staff feels that it is still somewhat premature to base source term calculations in a proceeding such as this upon them.

The above examples illustrate the staff's difficulty with the method used and some of the conclusions reached by Stratton, Rodger and Potter. We believe that quantification of "reduction factors" to be applied to source term estimates is premature.

The NRC research program is actively engaged at this time in developing substantial new information regarding severe accident source terms for light water reactors. The approach by which this is being done is rather complex; it involves the development and application of new computer codes to describe the important processes in the core region, the reactor coolant system and the containment during the degradation and melting of the core. In addition to code development, extensive experimental work is in progress here in the United States, and in foreign countries as well, to augment the data base for the verification and validation of these codes.

Using the codes developed for this purpose, a series of U.S. light water reactors are being studied, one at a time, taking significantly different accident sequences and carefully analyzing the release and transport of radioactivity during those sequences. For example, the first plant studied in this manner, the Surry PWR in Virginia, is being studied for the accident sequences large break loss-of-coolant-accident, small break loss-of-coolant-accident, station blackout, and what is called in the Reactor Safety Study, Event V, the interfacing system LOCA where the reactor coolant system ruptures directly outside the reactor containment building. After these selected sequences are analyzed, a careful appraisal of the dominant accident sequences for a plant of this type can be conducted to appraise the expected accident source terms for all of the dominant accident sequences, and thereby develop the compendium of source terms which describes the risk characteristics of that plant. In addition, a report describing the technical base for these computer analyses is being developed on a parallel schedule. In addition to the NRC-sponsored work, we expect to see published this coming summer the results of similar work by the industry degraded core group (IDCOR) as well as an interim report on the subject from an American Nuclear Society special committee chaired by Dr. Stratton.

Later this year, after it has undergone peer review, the staff of the NRC will appraise this technical data to determine what substantive changes in accident source terms may be justified at this time and will then advise the Commission of the significance of these results as well as proposing to the Commission what regulatory or policy changes might be appropriate based on these accident source term changes. Therefore, we believe that at this time it is premature to attempt a quantitative reassessment or restatement of accident source terms here in the Indian Point proceeding or in other cases as well.

STATEMENT OF QUALIFICATIONS
ROBERT M. BERNERO

I am Robert M. Bernero, Director of the Accident Source Term Program Office in the Office of Nuclear Regulatory Research at the U.S. Nuclear Regulatory Commission.

I hold the degrees of Bachelor of Arts (1952) from St. Mary of the Lake (Mundelein, Illinois), Bachelor of Chemical Engineering (1959) from the University of Illinois, and Master of Chemical Engineering (1961) from Rensselaer Polytechnic Institute.

Early in my technical career I was employed by the General Electric Company at the Knolls Atomic Power Laboratory from 1959 to 1966 where I worked on the design, construction, and test of pressurized water reactors for naval propulsion plants. In that work I gained substantial experience in reactor electrical and fluid systems as well as the chemistry and radiochemistry of reactor cooling systems.

From 1966 to 1972, while still with the General Electric Company, I worked at the Valley Forge Space Center where I participated in the development of nuclear power devices for space applications. In my final position there I served as Manager of Energy Conversion Engineering where I directed the design and development of a high temperature thermoelectric power system using silicon-germanium alloys.

In 1972 I joined the Atomic Energy Commission regulatory staff as a licensing project manager. From 1972 until 1975 I managed reactor licensing cases including pressurized water reactors and the Clinch River Breeder Reactor. In 1974, when the draft version of the Reactor Safety Study (WASH-1400) was published, I was a member of the NRC staff team which performed an in-depth review of that benchmark risk assessment.

From 1975 until 1977 I worked in the NRC's Office of Nuclear Materials Safety and Safeguards as the licensing manager of the Barnwell Nuclear Fuel Plant and as Chief of the Fuel Reprocessing and Recycle Branch.

From 1977 to 1979 I served in NRC's Office of Standards Development as Assistant Director for Material Safety Standards. After the Three Mile Island accident, I served in the staff of the NRC's TMI Special Inquiry. At the end of 1979 I was appointed to be the Director of the Probabilistic Analysis Staff in NRC's Office of Research. That group, now known as the Division of Risk Analysis, has long been the center for the development of risk analysis methods at the NRC.

My permanent position at the NRC is still Director, Division of Risk Analysis. However, in January of 1983 I was appointed to be Director of the Accident Source Term Program Office. This program office was formed in January to assure that source term related research results are implemented in policy and regulatory practice in a timely manner.