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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

April 24, 1992

MEMORANDUM FOR: The Chairman

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: STATUS REPORT ON CHANGES IN THE PROBABILITY OF A CORE DAMAGE ACCIDENT AS INFERRED FROM ACTUAL EVENTS OCCURRING BETWEEN THE 1970'S AND TODAY

The enclosed staff report on changes in the estimated probability of a core damage accident provides a response to the second part of your request dated October 28, 1991. A report on the first part of your request, regarding plant operation and maintenance costs, has been sent separately in a memorandum dated December 19, 1991.

The staff analysis of actual events indicates that the trend in the sum of estimated conditional core damage probabilities shows an overall decrease since the 1970's. This observation can be used as an indicator of a corresponding downward trend in the probability of occurrence of a core damage accident. While the staff believes that the trends and their relative changes over the years are correct, the absolute magnitude of the inferred core damage frequency has large uncertainties due to the following limitations in the ASP methodology:

(a) Not all plants have a probabilistic risk assessment (PRA), and therefore generic event trees and generic fault trees must be used.

(b) Plant-specific equipment failure rates and human error rates generally are not available, and therefore generic data must be used.

(c) External events, such as seismic events, floods, and high winds are not considered in the data.

(d) Accidents during shutdown that could lead to core damage were not evaluated until recent years.

Contributing to the downward trend are the many different regulatory and industry actions, addressing improvements in operator training, and in equipment performance and operations that have been taken since the Three Mile Island accident. The enclosed report briefly summarizes some of the key actions that we believe are responsible collectively for the observed decrease. However, additional study is required to quantify the significance of and to determine the degree of correlation with specific causes of this trend.

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The ACRS reviewed an early version of this report as indicated in its letter dated February 14, 1992; valuable comments were provided and factored into this report. The ACRS also provided comments on the ASP program in general and expressed the belief that additional valuable information may be contained in the precursor data. We agree and note that there are both ongoing and planned efforts to improve the individual estimates of event conditional core damage probability, better interpret the results, and evaluate the underlying causes of the observed trends. We plan to continue interacting with the ACRS and provide periodic reports to the Commission on the Accident Sequence Precursor Program.

Original Signed By:
James M. Taylor

James M. Taylor
Executive Director
for Operations

Enclosure: As stated

cc: Commissioner Rogers
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CHANGES IN PROBABILITY OF CORE DAMAGE ACCIDENTS INFERRED ON THE BASIS OF ACTUAL EVENTS.

I. OBSERVED CHANGES IN CORE DAMAGE PROBABILITIES

Nuclear power plant events identified as precursors to various core damage sequences have been analyzed and the conditional core damage probability of each event leading to core damage has been estimated. Figure 1 shows the annual summed conditional core damage probabilities[1] corresponding to the precursor event data that the staff analyzed (References 1 through 3) for 1969-1990 (data analysis for 1982-1983 are not yet available). It should be noted that some of the event data for the earlier years have been requantified on the basis of more current knowledge and event modeling methods (Reference 4). However, although the revised data would be expected to change some of the estimated conditional core damage probabilities, they would not affect the major features of the trends discussed below.

Since the amount of operating experience (in reactor years) varies among calendar years, the core damage index (CDI) can represent the trend better. The CDI is calculated by dividing the sum of the conditional core damage probabilities for each calendar year by the number of reactor years of experience for that year. The resulting CDI is a core damage measure that represents an "average" throughout the industry. Figure 2 shows the trend of the CDI for the data in Figure 1.

The 1970's saw a number of events with significant conditional core damage probabilities. For example, the Browns Ferry fire represents 93% of the annual sum of conditional core damage probabilities for 1975. Similarly, the Rancho Seco event involving loss of Auxiliary Feedwater flow contributed 86% to the annual sum in 1978. For 1979, the accident at Three Mile Island (where core damage occurred) contributed essentially all of the annual sum, even though 34 precursors were identified for that year. These types of events dominated the annual contributions to the risk of core damage during the 1970's.

In the 1980's, events of similar risk significance did not occur. Hence, the sum of conditional core damage probabilities decreased substantially. It is noteworthy that even the more significant precursors in the 1980's, such as the Davis-Besse and Vogtle events, did not have the high conditional core damage probabilities seen in

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[1]It should be noted that the annual sum of conditional core damage probabilities is not a probability. Hence, it can exceed the value of one (for 1979 it is 1.05). The information content within the sum includes both precursor frequency and core damage likelihood. This is a limitation in that it does not distinguish, for example, between many "low" and one or two "high" contributors to core damage probability. Nevertheless, the sum is a convenient indicator of the collective plant safety performance of operating plants for any given year. The term "conditional" refers to the fact that the probability of core damage for an event is estimated given that the precursor event has occurred.

the 1970's. Overall, both figures indicate that the probability of core damage has decreased substantially since the 1970's.

One way to interpret the precursor data is to consider separately two independent aspects of the likelihood of core damage. One is event severity. This is dependent on the nature of the initial hardware failure or operational error, as well as plant readiness and ability to control and recover from the event. Mitigation and recovery actions can play an important role in reducing core damage probability. The other factor is the frequency of initiating events. A higher frequency of initiating events provides more opportunities for sustaining an event that can lead to core damage. Typically, both factors affect the overall likelihood of a core damage accident.

EVENT SEVERITY

Figure 6 shows the distribution of the most significant events by year (i.e., those with conditional probabilities greater than 1.0E-1). Events with lesser conditional probabilities (in the range of 1.0E-2 to 1.0E-1) have a similar distribution, as shown in Figure 7. Clearly, there is a general absence of events with relatively high conditional core damage probabilities in the 1980's.

For events with still lower conditional core damage probabilities (Figures 8 and 9), the decreasing trend is no longer evident. That is, precursor events with lower conditional core damage probabilities still keep occurring at a relatively constant rate. Overall, it is difficult to make any causal interpretations with respect to the distribution of lower ranges of conditional core damage probabilities shown in Figures 8 and 9. At the very least, interpretation is hampered by changing reporting requirements and changing ASP models within the period between 1969 and today.

However, it is clear that the results from the 1970's are dominated by a limited number of very risk significant events which have not been observed in the 1980's. Changing reporting requirements and ASP models are not likely to impact this.

EVENT FREQUENCY

Scram frequency can be used as one measure of initiating event frequency. The general trend in scram reduction in recent years can be seen in Figure 3. The data in Figure 3 show the number of plant trips per year from 1984 through 1990. As indicated, the number of trips per year has been decreasing continuously. Data for the period 1984-1987 show that the principal causes for unplanned scrams are equipment malfunctions (~61%), human error (~25%), and faulty procedures (~6.5%).

Figure 1 reflects this with respect to similar changes in the sum of conditional core damage probabilities over the years 1984 to 1990. It is interesting to note that there is a significant correlation between reactor trips and the sum of conditional core damage probabilities for the period 1984 to 1990 (as shown in Figure 4). An even greater correlation exists with reactor trips with complications (e.g., transient with trip and subsequent failure of standby equipment) as shown in Figure 5.

Similar comparisons for earlier years, (especially prior to 1984) cannot be made as easily. The data needed to do this are not readily available and would require a significant data acquisition and analysis effort. We believe such effort is not warranted because the precursor data from earlier periods appear to be dominated by factors other than numbers of plant trips, as discussed previously.

II. REGULATORY IMPROVEMENTS

While recognizing that it is not possible to prove a causal relationship, the staff believes the reduction of significant events in the 1980's can be attributed in large measure to regulatory actions directly related to improvements in plant equipment and operations. In particular, regulatory requirements stemming from the TMI Action Plan (Reference 5) led to improvements that provided better control of event initiation, mitigation, and recovery, which contributed to the general reduction of conditional core damage probabilities since the 1970's. Two examples are the following:

- Auxiliary Feedwater (AFW) improvements

  The requirements of the TMI Action Plan (Reference 5) led to a number of improvements in the AFW system and its initiation. These include generic and plant specific modifications to Technical Specifications, water supplies, piping arrangements, instrumentation, and safety-grade automatic initiation. The latter has been implemented at all PWR plants (except Yankee Rowe). Other improvements include physical and procedural changes for reducing the potential for steam binding of AFW pumps, as well as the provision of redundancy and diversity.

  - Due to these improvements, including the complete independence of at least one train of the AFW from all A-C power supplies, the reliability of the AFW system has increased significantly, ranging from a factor of 5 to as much as 100. The AFW system plays a key role in the mitigation of most core damage accident sequences. Hence, its improved reliability can be credited as one of the principal factors behind the observed reduction in the likelihood of core damage.

- Reactor Coolant Pump (RCP) seals

  The failure of the RCP seals can lead to a seal LOCA, an accident sequence which has been identified as a significant contributor to the probability of core damage. In the 1970's, one important seal failure mode was the potential loss of seal cooling in the event of a loss of offsite power (LOOP). As part of the TMI Action Plan (Reference 5), a requirement was made to reconfigure the power supply so that, in the event of a LOOP, emergency diesel power was available for operating seal cooling equipment. Considering that a LOOP occurs relatively often, the requirement for providing reliable power to seal cooling is viewed as a significant contributor to the reduction of the likelihood of core damage due to a seal LOCA.

Other regulatory actions led to major improvements in the areas of human factors and plant operations. These also played a key role in reducing the occurrence of significant events. These principal regulatory actions addressed control room human factors, symptom-based emergency procedures, and operator training.

- Control Room Human Factors Improvements

  In the area of human factors, the requirements of the TMI Action Plan (Reference 5) led to two major initiatives directed at improving operator performance and safety at nuclear power plants. These were the Detailed Control Room Design Review (DCRDR) and the Safety Parameter Display System (SPDS). The DCRDR led to improvements that provide better information and enhance the ability of control room operators to prevent or cope with accidents. The SPDS provides plant operators at all plants with simple, continuous, and integrated indication of plant safety status. The implementation of both the DCRDR and the SPDS has improved operator performance during abnormal events occurring at plants.

- Symptom-based Emergency Operating Procedures (EOPs)

  In response to the TMI accident, the NRC delineated requirements for improved EOPs (Reference 5). The licensees were required to develop function-oriented (symptom-based) EOPs which would be easier to use by plant operators. These EOPs were intended to enable the operating crew to mitigate the consequences of a broad range of initiating events and subsequent multiple failures, including operator errors, without the need for initial diagnosis of the events. Specific improvements associated with function-oriented EOPs include

1) Enhanced crew awareness of plant conditions,

2) Improved crew ability to deal with challenges to plant integrity,

3) Identification of a preferable method of recovery from transients, and

4) Reduced likelihood of event misdiagnosis leading to severe plant degradation.

- Improved Operator Training Using Plant Simulators

  As mandated by the 1987 revision to the 10 CFR 55, all licensed operator initial and requalification operating tests administered after May 26, 1991 are to be conducted, in part, on a simulation facility that was either approved by the Commission or certified by the facility licensee. The revised rule contributed to the significant increase in industry resources committed to simulator design, construction, and improvement. This led to notable advancements in simulator fidelity and capability and promoted the widespread use of simulators (all but one plant) for operator training and examination. This, in conjunction with NRC administration of requalification examinations, has resulted in improvements in training and testing of operators. Hence, currently, nuclear power plant operators are provided with more realistic emergency and abnormal event response training and examinations that are more operationally oriented.

III. REGULATORY OVERSIGHT

In the aftermath of the Davis-Besse event in June 1985, the NRC staff took a closer look at its procedures for evaluating the operational safety performance of power reactor licensees. The staff examined its inspection program, the Systematic Assessment of Licensee Performance (SALP) program, and its own internal processes for integrating all agency information available on each licensee's safety performance.

The conclusion from that introspective look was that NRC should focus more effort on improving operational safety and should improve its method of integrating the information gathered from inspections, licensing reviews, SALP reports, operational data analyses, and special safety analyses, to arrive at an overall assessment of how safely licensees were operating their plants. Accordingly, the staff implemented semi-annual meetings of senior headquarters and regional managers to review overall safety performance at operating plants. From these senior manager reviews, the staff has developed a "Watch List" of plants which, because of their poor operational safety performance, warrant increased NRC attention.

In addition to improvements in the methods for integrating all information on plant performance, it became clear that NRC's inspection program would have to focus more sharply on assessing how well plants were being operated. As a result, the inspection program in recent years has become more diagnostic in nature, and not merely compliance-oriented as in past years. This change in focus has been accompanied by an increase in the number of team inspections and a corresponding evolution in SALP reports toward more critical assessments of the safety management of nuclear plants. The SALP program is particularly effective in providing a means for the staff to evaluate a licensee's performance and to examine the underlying causes for operational problems. It further provides an agenda for professional, non-adversarial discussions between senior NRC managers and senior utility executives regarding areas for improving operational safety performance.

In judging the net result of this increased regulatory oversight focus on operational safety, the staff concludes that there has been a significant improvement in the overall safety of operating plants in recent years. The extent to which this improvement can be attributed to NRC actions, or INPO actions or individual utility initiatives, cannot be measured. Nonetheless, the staff believes that the enhanced regulatory oversight process of identifying declining performance and requiring improvements in operational safety is a major reason for the absence of high conditional core damage probability events today relative to the 1970's.

IV. INDUSTRY ACTIONS

Closely related to regulatory actions are various industry efforts that have been directed at improving the safety of nuclear power plants since the 1970's. Some of these include

- Plant trip reductions.

  In 1984 the Nuclear Utility Management and Human Resources Committee (NUMARC) established an industry average performance goal for unplanned scrams for 1985. The goal was to have no more than three unplanned automatic scrams per unit per year (while

Subsequently, at the request of the industry, the Institute of Nuclear Power Operations (INPO) tracked scrams, analyzed their causes, and recommended methods for reducing the frequency of unplanned scrams. In particular, INPO formulated a number of recommended good practices in the areas of design, equipment, maintenance, testing, and operation that were aimed at reducing the number of unplanned scrams.

Additional study by INPO for the three year period 1984-1986 led to the establishment of longer-term industry performance goals for unplanned scrams. Scram reduction programs were established by each of the Nuclear Steam Supply System (NSSS) vendor owners groups in an effort to reach the industry goals. The result was that the average automatic scram rate for 1990 dropped to about 1.6, which compares well with the industry goal of 1.5 for that year.

The principal reasons for the observed reductions in plant trip rates include equipment modifications, improved preventive and corrective maintenance practices, and generally a higher sensitivity to unplanned reactor trips by the industry as a whole.

- INPO Plant Evaluations

  INPO plant evaluations have been a significant factor in providing utilities with critical peer evaluations of their plant operational practices and, perhaps more important, in providing a means for promulgating good operating practices throughout the utility industry.

- INPO-training accreditation program.

- INPO Significant Event Evaluation and Information Network (SEE-IN) Program.

V. FUTURE ACTIVITIES

Efforts to improve the ASP program and to make better use of the results fall into three categories as discussed below. These include both ongoing and planned efforts.

ASP Model Improvement - Efforts are under way to improve the estimates of individual event conditional core damage probability. While ASP models will likely remain less complex than typical PRA models, they will be checked for consistency with state-of-the-art PRA models and, within the limitations of a simpler model, be brought up to the standards of typical PRAs. The ability to perform uncertainty evaluations will also be included. In addition, the models will be made more plant-specific.

INTERPRETATION OF ASP RESULTS - Efforts are also under way to improve the evaluation of the implications of overall ASP results, to improve the display of the results, and to determine the statistical significance of apparent trends. A key effort will involve determining if the observed precursors from past events can be used to estimate the actual overall industry risk.

EVALUATION OF ASP RESULTS - Efforts to evaluate trends in ASP results and the underlying causes will continue.

VI. SUMMARY

The observed trends for 1969-1990 indicate a decrease in the sum of conditional core damage probability. This sum depends on the frequency of precursor events and the severity of conditional core damage probabilities. The decline in the sum of conditional core damage probabilities since the 1970's is the combined result of several major regulatory actions as well as industry efforts taken to deal with the causes of major precursors observed in the 1970's. Other regulatory and industry actions also may have contributed to the decline, but would be much more difficult to correlate. Several efforts are under way to provide additional insights stemming from existing ASP data.

VI. REFERENCES

1. "Precursors to Potential Severe Core Damage Accidents.", Status report for the years 1969 through 1979, NUREG/CR-2497, Volumes 1 & 2.

2. "Precursors to Potential Severe Core Damage Accidents.", Status report for the years 1980 through 1981, NUREG/CR-3591, Volumes 1 & 2.

3. "Precursors to Potential Severe Core Damage Accidents.", Status report for the years 1984 through 1990, NUREG/CR-4674, Volumes 1 through 14.

4. "A Review of Comments on the 1969-1979 Accident Sequence Precursor Program Status Report; NUREG/CR-2497", Technical Letter Report, ORNL/NRC/LTR-85/14

5. TMI Action Plan, NUREG-0737, Supplement 1.

FIGURE 1. ANNUALLY SUMMED ASP CONDITIONAL CORE DAMAGE PROBABILITIES

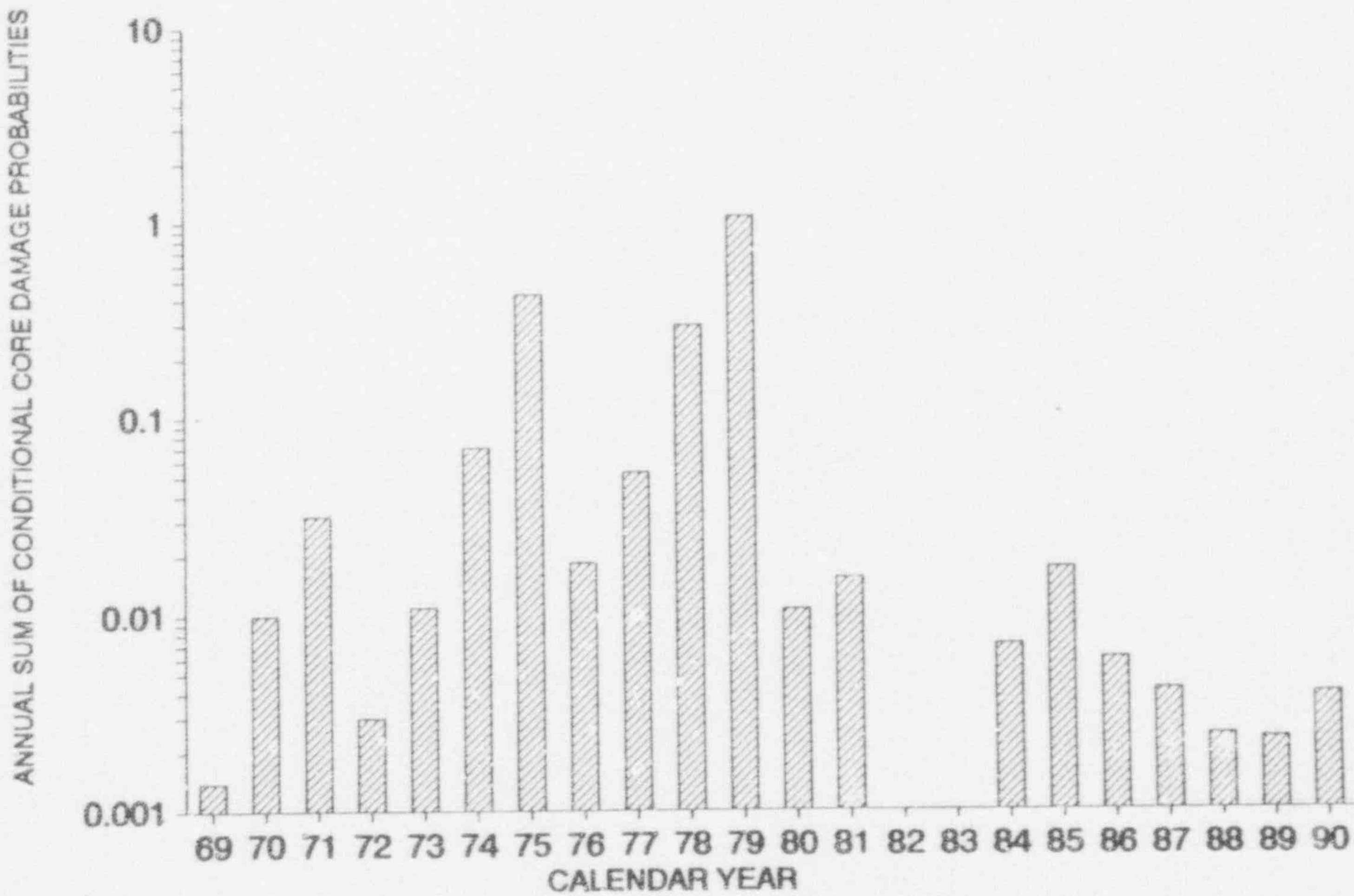


FIGURE 2. ANNUAL CORE DAMAGE INDICES

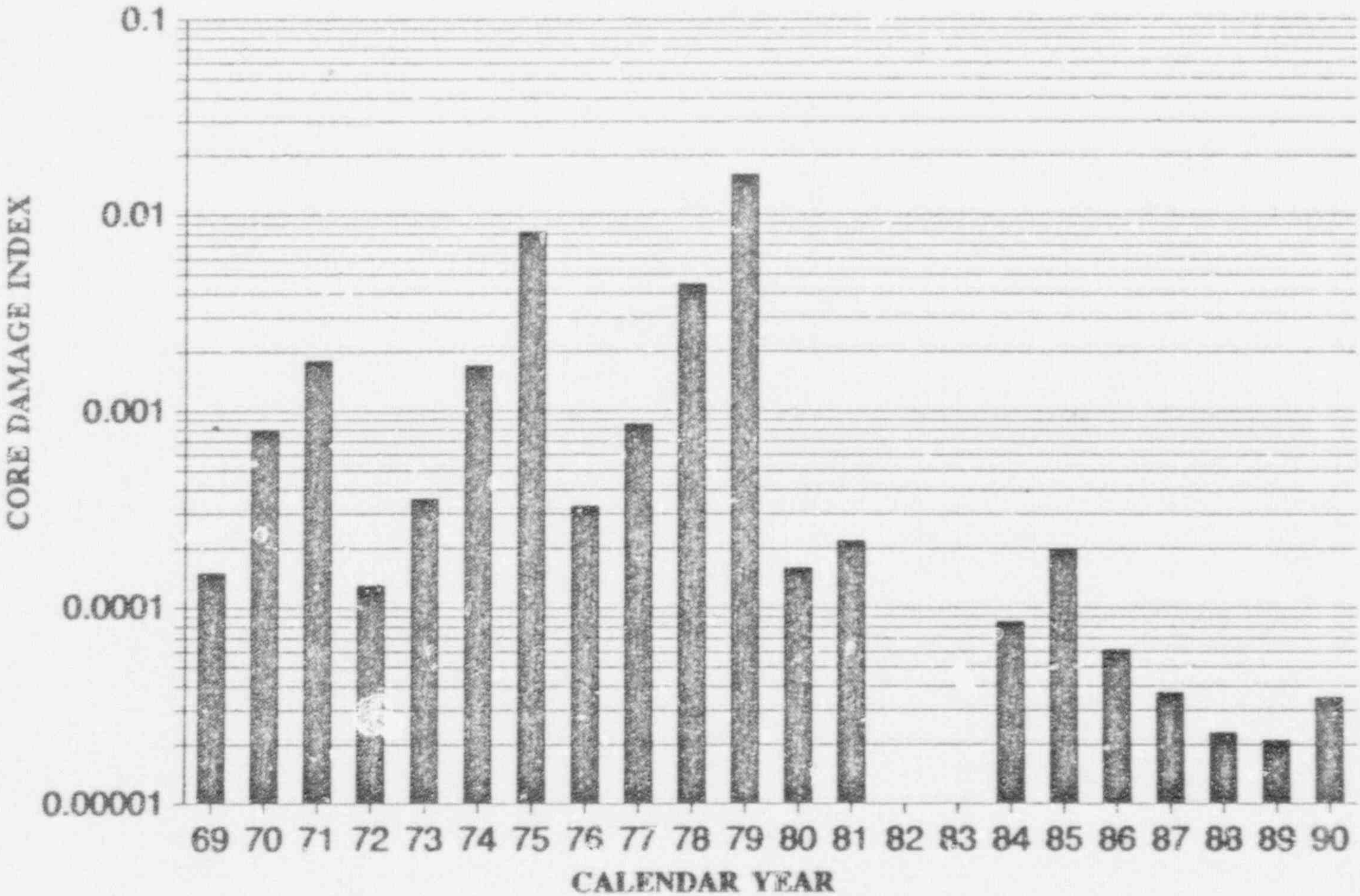


FIGURE 3. MANUAL AND AUTOMATIC REACTOR SCRAMS FROM CRITICALITY

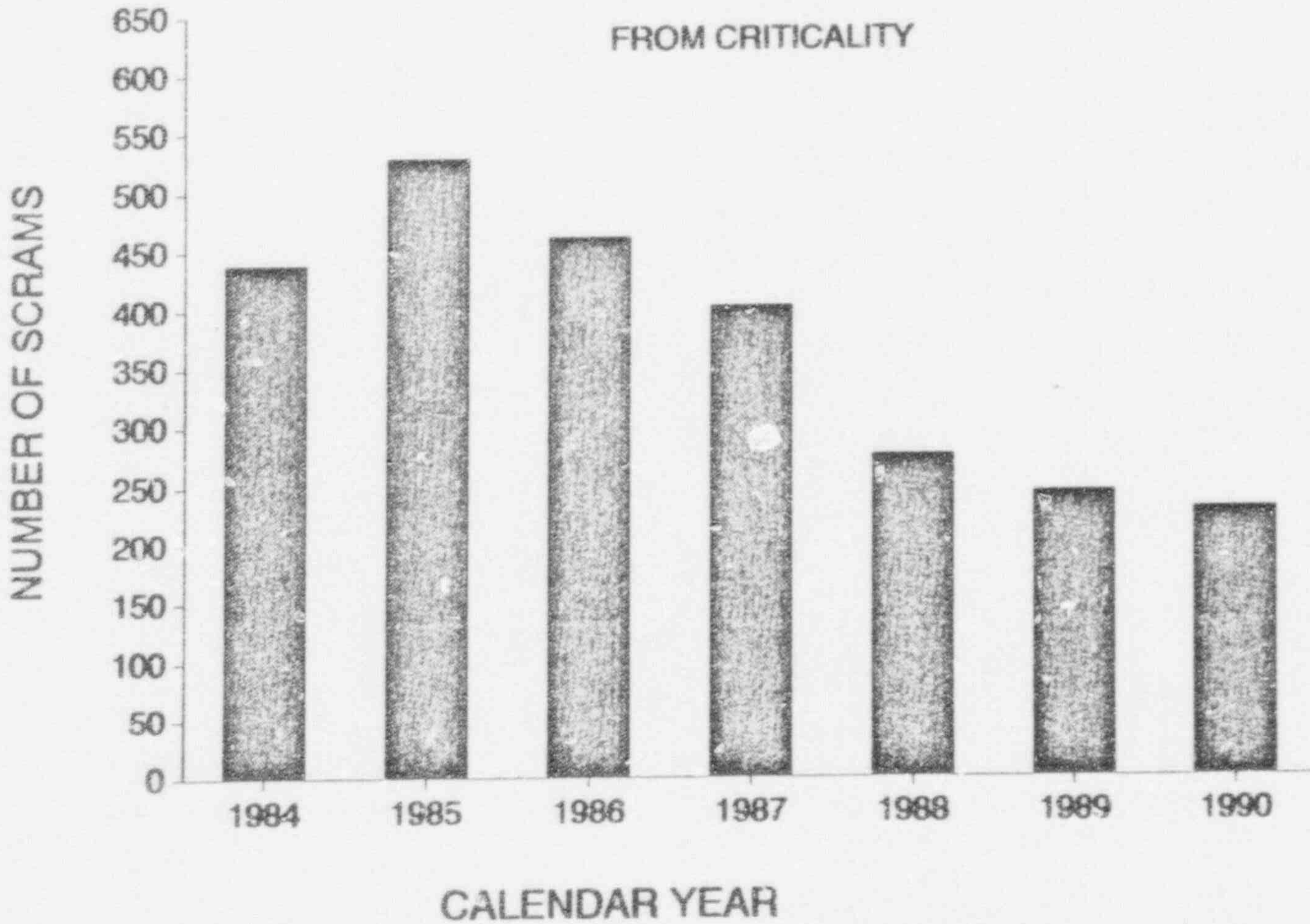


FIGURE 4. CORRELATION OF ANNUALLY SUMMED CONDITIONAL CORE DAMAGE PROBABILITIES (CCDP) AND REACTOR TRIPS

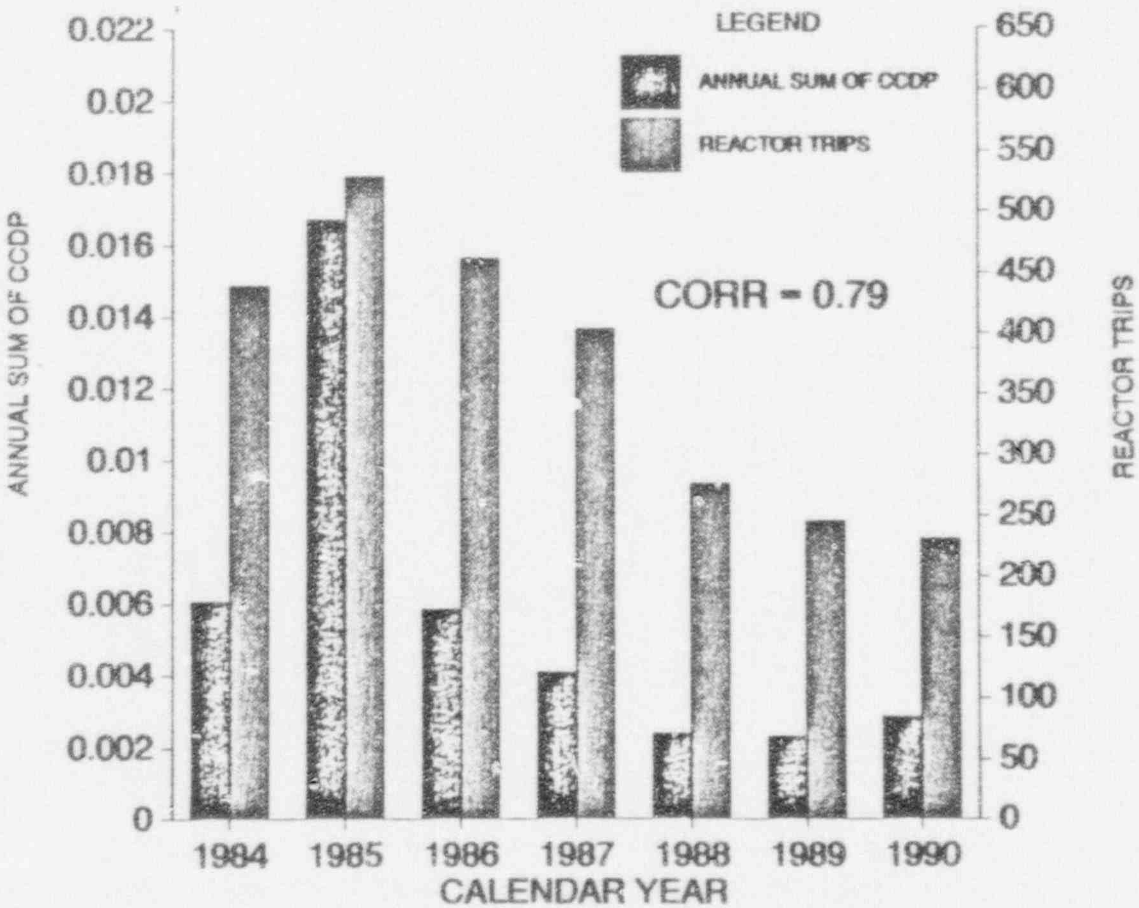
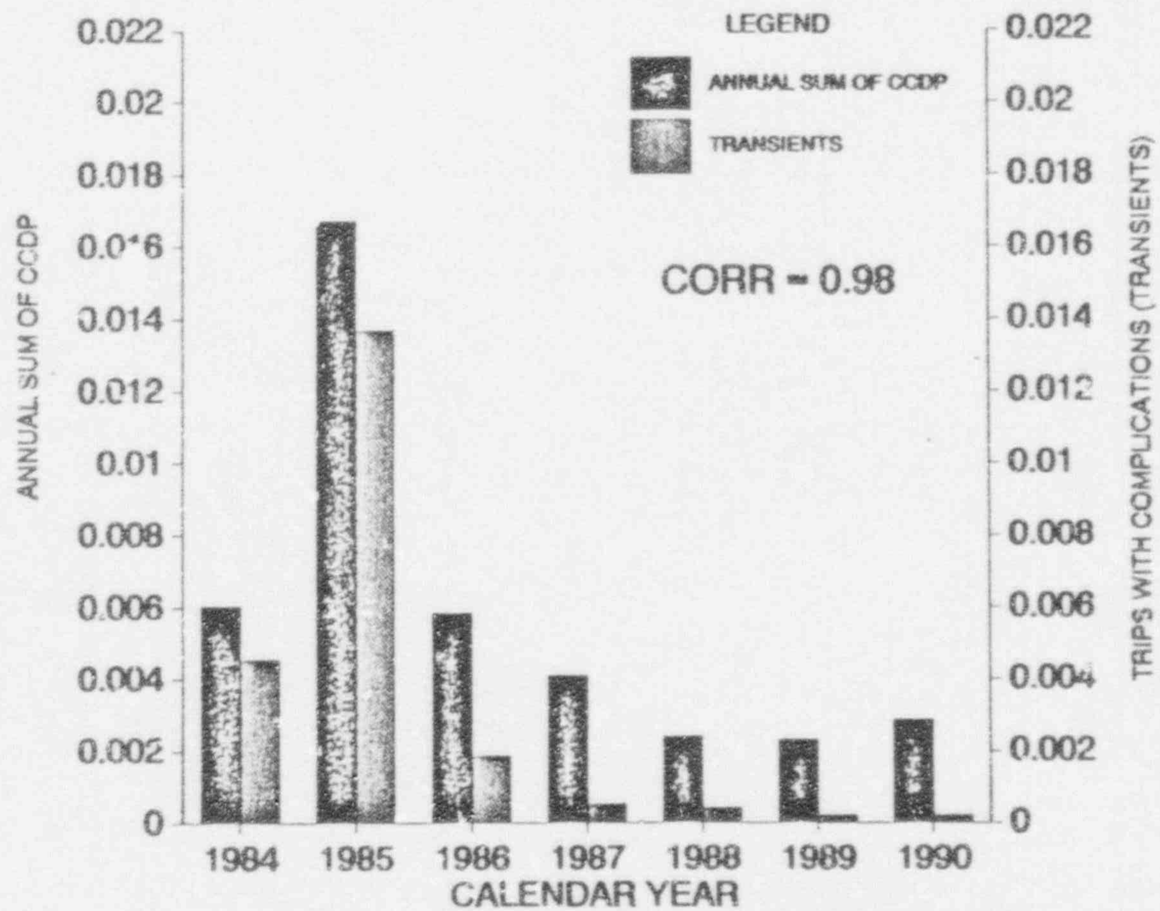


FIGURE 5. CORRELATION OF ANNUALLY SUMMED CONDITIONAL CORE DAMAGE PROBABILITIES (CCDP) AND TRANSIENTS



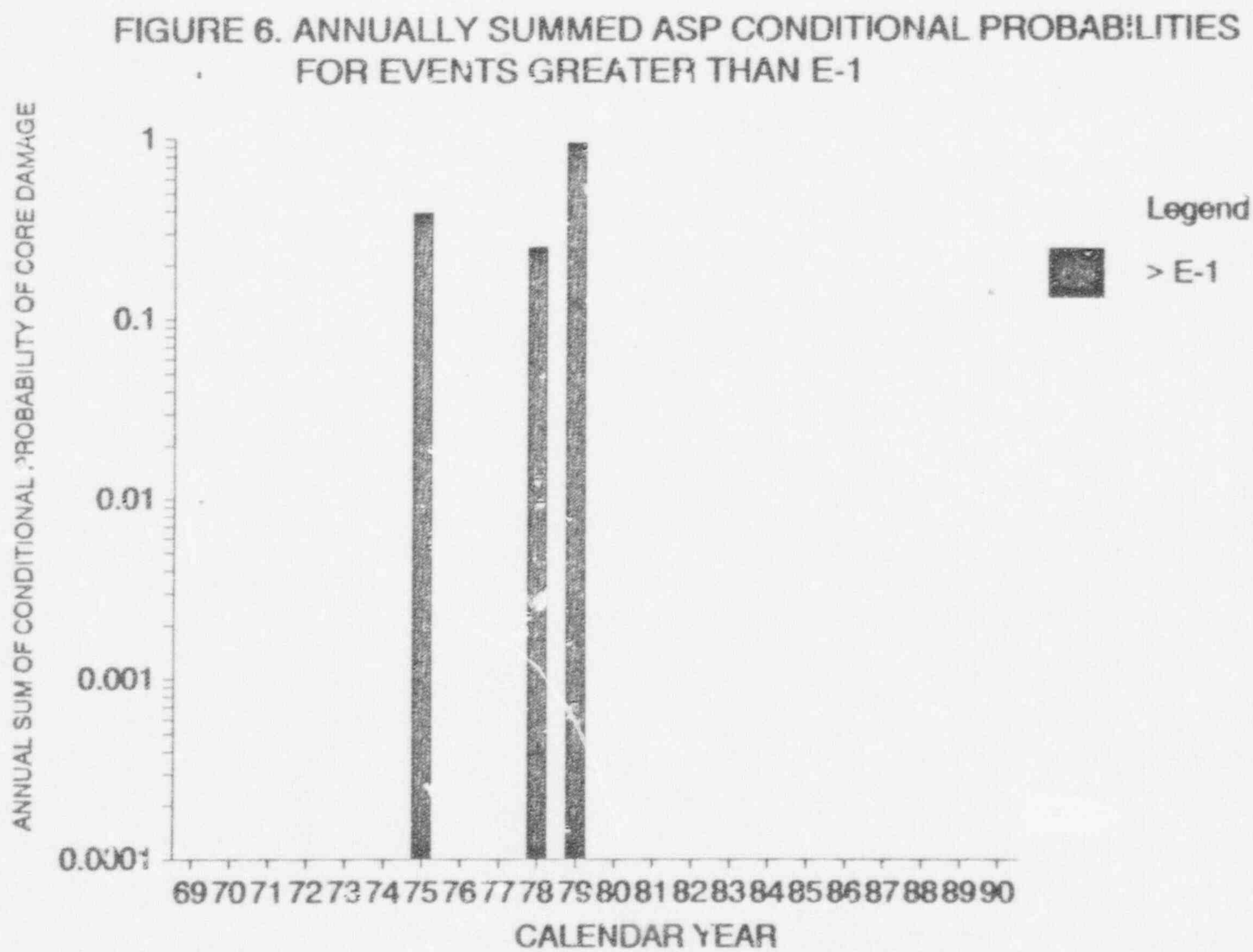


FIGURE 6. ANNUALLY SUMMED ASP CONDITIONAL PROBABILITIES
FOR EVENTS GREATER THAN E-1
ANNUAL SUM OF CONDITIONAL PROBABILITY OF CORE DAMAGE
1
0.1
0.01
0.001
0.0001
69 70 71 72 73 74 75 76 77 78 79 80 81 82 83 84 85 86 87 88 89 90
CALENDAR YEAR
Legend
> E-1

FIGURE 7. ANNUALLY SUMMED ASP CONDITIONAL PROBABILITIES FOR EVENTS E-2 TO E-1

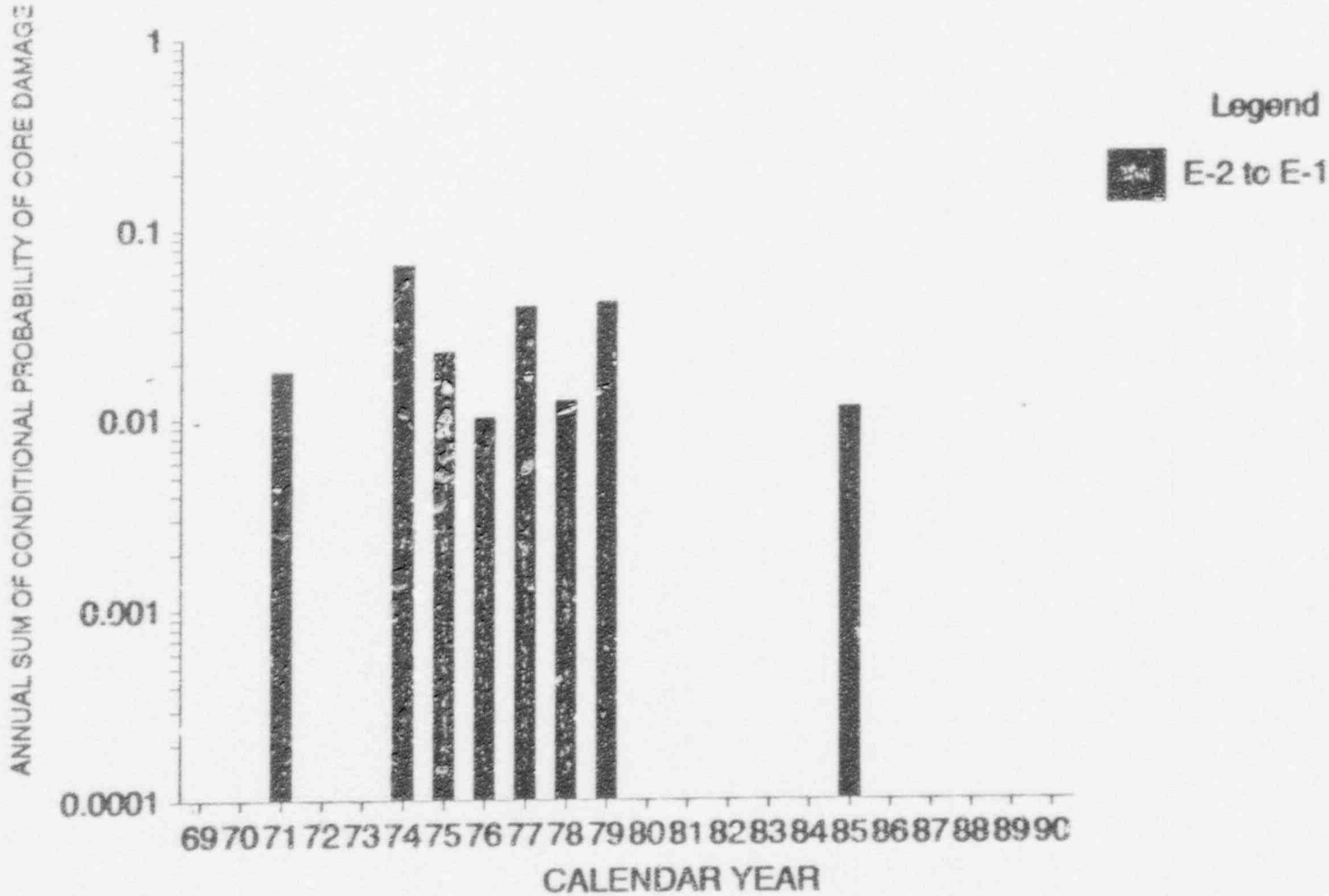


FIGURE 8. ANNUALLY SUMMED ASP CONDITIONAL PROBABILITIES FOR EVENTS E-3 TO E-2

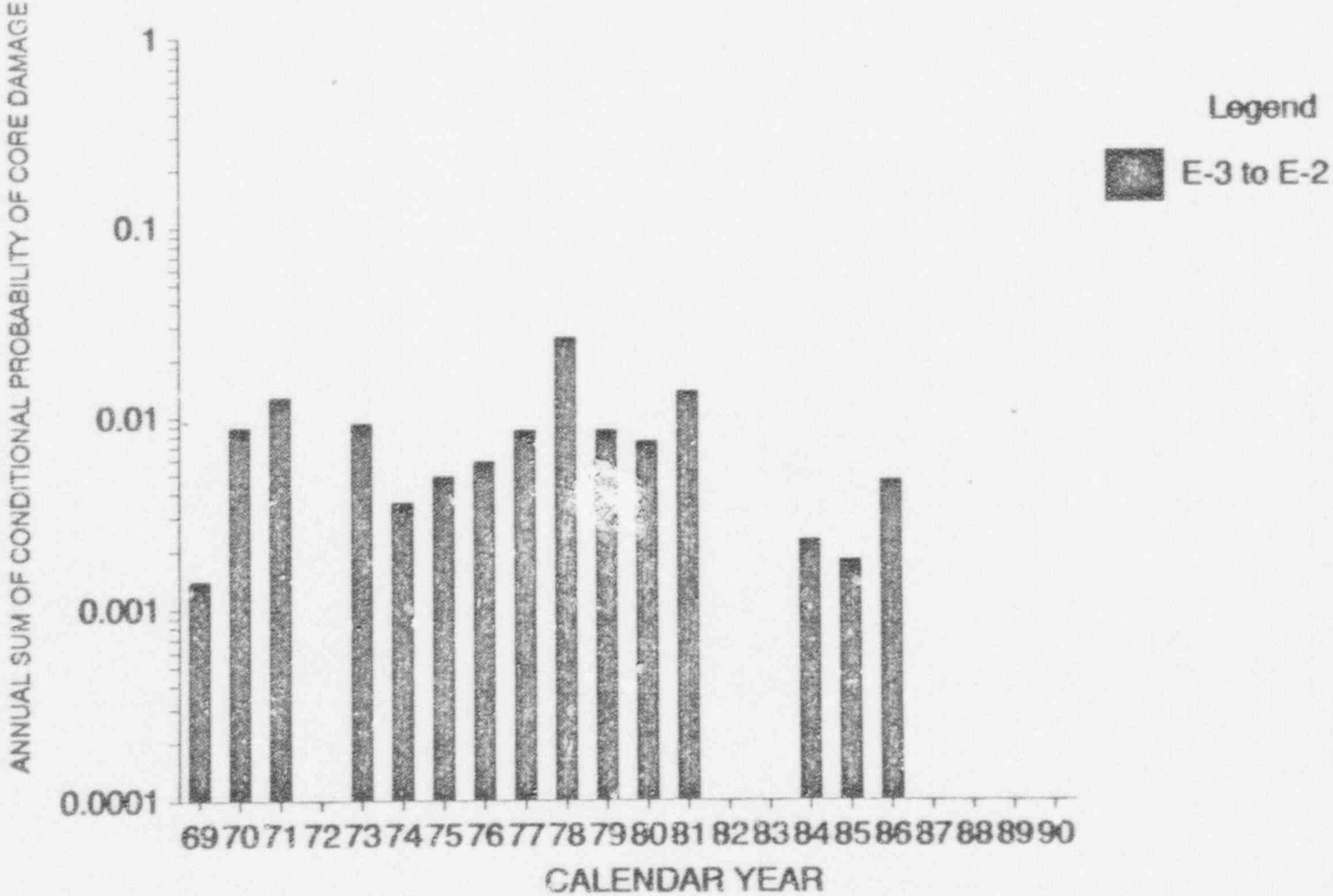
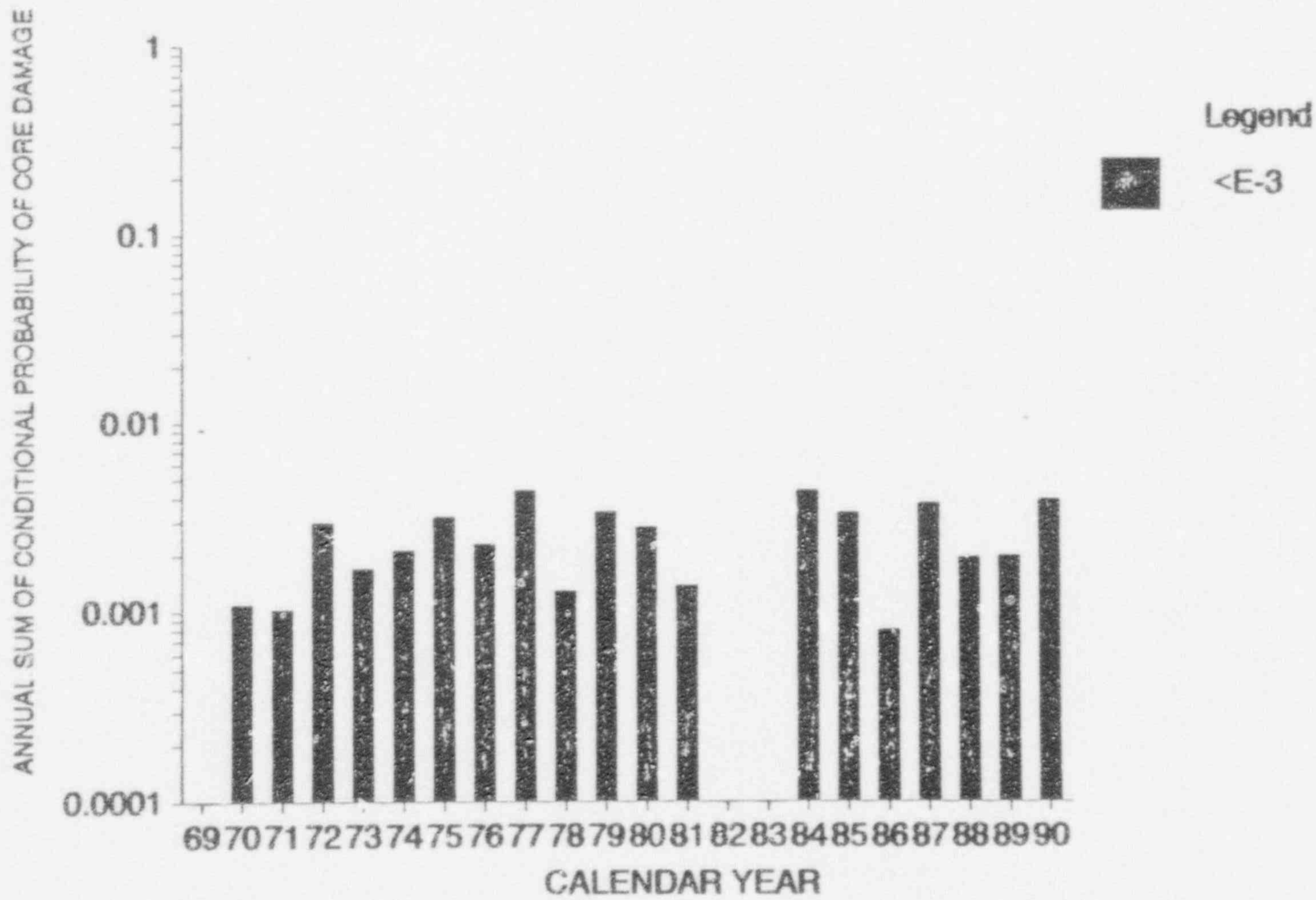


FIGURE 9. ANNUALLY SUMMED ASP CONDITIONAL PROBABILITIES
FOR EVENTS LESS THAN E-3



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CHAIRMAN

October 28 , 1991

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Ivan Selin

SUBJECT: REQUEST FOR ANALYSIS

Since becoming Chairman in July, I have visited a dozen nuclear power plant sites in the U.S.. A number of the licensees for these plants have expressed concern about substantially increased operation and maintenance costs since the 1970's. I would appreciate your insights into the reasons for the increased operation and maintenance costs. I would also be interested in your analysis of the change in the probability of a core melt accident based on actual events from the 1970's through today and the reasons for the change. I believe it would be beneficial if your analysis received peer review from one of our advisory committees, such as the Advisory Committee on Reactor Safeguards.

cc: Commissioner Rogers
Commissioner Curtiss
Commissioner Remick
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