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November 11, 1992
ST-HL-AE-4261
File No.: G31.03

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Request for Additional Information Regarding
Review of the Proposed Changes to the South Texas Project
Technical Specifications (TAC NOS. M76048 and M76049)

Reference: 1) Letter dated August 18, 1992 from George F. Dick to Mr. Donald P. Hall, "Request for Additional Information Regarding Review of the Proposed Changes to the South Texas Project Technical Specifications (TAC NOS. M76048 and M76049)"

In accordance with the request made by Reference 1, enclosed with this letter (Attachment 1) are responses to the questions provided regarding the proposed South Texas Project Technical Specification changes currently under review by the NRC and its contractor, Brookhaven National Laboratory (BNL).

If you should have any questions, please contact A. W. Harrison at (512) 972-7298 or me at (512) 972-7205.

William J. Jump
William J. Jump
General Manager,
Nuclear Licensing

SDP/ag

Attachment 1: Responses to the NRC Regarding the Risk-Based Evaluation of STPEGS Technical Specifications

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Revised 10/11/91

Attachment 1
Responses to the NRC Regarding the Risk-Based Evaluation of STPEGS Technical Specifications

1. Provide a copy of the revised system models and the plant models (accident sequence models) which were developed using the RISKMAN® code, Version 3.06 as part of the development of the bases for the proposed changes to the current Technical Specifications (TS). A schedule and mechanism for the transfer of the model can be arranged during future discussions. Your initial response should address if you are agreeable to supplying your model and what restrictions you desire to impose if the model is provided. Also provide a list of the assumptions used in quantifying the risk impact of the proposed changes to the allowed outage time (AOT) and surveillance testing intervals (STI).

RESPONSE:

HL&P considers the STP risk model privileged and proprietary, and, of course, the RISKMAN® PC software is proprietary to PLG, Inc. The STP risk model files are embedded in the RISKMAN® software, and these files would be of no use and unintelligible without this software. It is HL&P's understanding that Brookhaven National Laboratory will be performing the review of the technical basis for the proposed changes to STP technical specifications. Brookhaven National Laboratory (BNL) should obtain a license for the RISKMAN® code, Version 3.06 or later, from PLG, Inc.

HL&P will work with BNL to provide the information needed to assess the basis for the results of the "Risk-Based Evaluation for STPEGS Technical Specification," (ST-HL-AE-3283 dated February 1, 1990). Your representative should contact Mr. Richard P. Murphy at (512) 972-8919 or Mr. C. R. Grantom at (512) 972-7372 to discuss these requirements. As a condition for providing the STP plant model to BNL, it is requested that prior to publication of analyses, or evaluations for proposed actions, publication or other formal reports, HL&P should be given the opportunity to duplicate the evaluations and provide comments.

Please note that although the RISKMAN® model is complete, some of the new system fault trees do not include the definitions required to evaluate all of the proposed technical specification changes. This was done for simplicity and to expedite the IPE submittal, and does not affect the quantitative results. These definitions are currently being incorporated and are expected to be complete before the end of the year. In addition, several initiating events which are affected by the proposed technical specification changes, e.g., loss of EAB HVAC, are currently quantified separately for input to the RISKMAN® model. Conversion of these initiating events to the RISKMAN® format is underway. In addition, other options are available for quantification of these initiating events.

It is highly recommended that the NRC and BNL send 1-2 (preferably 2) analysts to STP for up to a week for familiarization with the STP model. The scope of familiarization would include a detailed review of the assumptions used in quantifying the risk impact of the proposed changes to the AOT and STI and the implementation of those assumptions in the model requantification. The STP model is very complex and the process of making the changes to the individual model and/or groups of systems models for each technical specification change, updating the database, and requantification of the event tree model should be well understood - beyond the level of contractor training on the RISKMAN® software with simplified files.

The "Risk-Based Evalua... of STP Technical Specifications" attached to ST-HL-AE-3283 dated February 1, 1990, identi...s all of the assumptions made in quantification of the risk model. The assumptions are few; the implementation of these assumptions is very complex and is documented in an 8 volume calculation. This process should be reviewed in detail by a BNL analyst at HL&P before attempting to requantify the STP RISKMAN® plant and systems models.

2. Provide a list of all major areas of the conservatisms that are in the current version of the South Texas PSA.

RESPONSE:

In general, the major assumptions and conservatisms used in the Level 1 South Texas PSA have been documented in NUREG\CR-5606. Particular attention should be paid to Sections 2.0 and 3.0 of this NUREG as it delineates most of the conservatisms and important assumptions used in the study. Follow-on actions described in Sections 2.0 and 3.0 have been addressed in the STPEGS Level 2 PSA and IPE which was submitted to the NRC on August 28, 1992 via ST-HL-AE-4193. Also, the Level 2 PSA and IPE contain conservatisms with respect to containment performance phenomenology; however, the assumptions and conservatisms used are consistent with those contained in NUREG-1150 and NUREG-4551. Also, the STPEGS Level 2 PSA and IPE contains important new software technology features which have reduced some modeling conservatisms to provide more accurate quantification of STP's core damage frequency (See Section 1.4.1). A few of the more notable conservatisms still existing in the PSA are:

- Accumulators - they are not assumed to prevent core damage but are modeled only to reflect proper plant damage state binning.

- Operator Actions - in general the values used to quantify success of human actions are greater than the generic values. This is due to the operator surveys which were performed early in the PSA at a time before STP commercial operation, and which reflected operator confidence of successfully performing selected human recovery actions at that time. Only a few of these operator actions have been updated as part of the IPE.

- The success criteria used in both the PSA and the IPE for EAB HVAC assumed that 450 tons of chilling were required during normal operation to provide safety related cooling in the Electrical Auxiliary Building (EAB). Actual operating experience has shown that only 300 tons of chilling capacity or less will remove normal heat loads.

- The success criteria used for safety injection under LLOCA conditions requires that two of three trains be operating for success. One safety injection train providing flow to a unbroken loop will provide adequate water to the RCS under LLOCA conditions.

- Use of the motor-driven startup feedwater pump to provide secondary side cooling is not included in the risk model.

- Use of the AFW system cross-over piping to provide flow to an intact steam generator is not included in the model.

- The steam dump valves to the condenser are not included in the risk model.

- V sequence is conservative since a detailed piping fragility calculation would most likely identify components of highest fragility to be inside containment (e.g., RHR heat exchangers, RHR pump seals).

Most of the above features were not included to facilitate ease of modeling in an already overly complex model which taxes the limits of software and hardware.

With respect to other assumptions used in quantifying STP risk, the following assumptions and conservatisms serve to highlight some of the key aspects of STPEGS risk model.

a. Preventative Maintenance (PM) activity durations are conservatively modeled in the Technical Specification (T/S) Study. Planned maintenance durations are modeled with a constant value of 60, 67 or 72 hours depending on the system. As part of the STP IPE effort and in conjunction with the upgrade of software to the new RISKMAN® 3.0 model, plant specific data was collected to develop a plant specific mean and distribution for the ECW system. The actual "as experienced" PM duration for the ECW (and SDG) system is now modeled by a distribution with a mean of approximately 39 hours. In this regard the previous model for ECW was extremely conservative as are all other systems where PM activity durations are modeled. Now the ECW conservatism, which was primarily due to lack of data, has been replaced by more accurate plant specific data. However, the PM durations for the remaining systems are still considered very conservatively modeled.

   It is for this reason that the plant specific PM durations, when available, for the remaining systems will be incorporated into STP RISKMAN® system models. This will remove the substantial conservatism presently reflected for planned maintenance outage durations. It should be noted that this is not to say that the effect of longer PM duration times (i.e., 60, 67, or 72 hours) won't be assumed or evaluated, for example, in sensitivity or other analyses; however, the approach taken for this study is to accurately reflect the plant specific operating experience, the STP maintenance philosophy and the STP work control practice. It should also be noted that the PM 12 week schedule is being substantially modified for some systems (e.g., ECW, SDG and AFW) to reduce the frequency from every twelve weeks to every 24 weeks or less (i.e., every refueling outage) with an outage duration of 48 hours. This will increase availability and reduce the annual outage duration even more.

b. The removal of some conservatisms with respect to the IPE risk model quantification is addressed in the STPEGS Level 2 PSA and IPE in Section 1.4.1 under the "Elimination of Support States" and "Updated Electric Power Recovery Analysis."

3. Provide a list of support systems and front line systems (by train) along with the time intervals allocated for the planned maintenance and testing events according to the 12 Week Maintenance Calendar established for the South Texas facility. If available, provide data on actual test and maintenance durations due to planned maintenance events, for which the safety system trains were unavailable.

RESPONSE:

Attached for your information are Tables 3.1 and 3.2 which list the support and frontline systems by train which are included in the 12 Week Maintenance Cycle. In general, systems are taken out of service on a quarterly basis; however, recent revisions to the Maintenance Cycle have been approved starting in 1993 which will result in semi-annual maintenance for Auxiliary Feedwater and Standby Diesel Generators and 18 month maintenance for High Head Safety Injection. This will improve availability; however, standby failures will need to be evaluated based on other non-component outage activities (e.g., predictive maintenance, increased walkdowns, increased sampling).

Actual test and maintenance data have been obtained for a 144 week history from the Operability Tracking Logs (OTLs) of both STP units. The OTLs are used to track equipment outages and their associated LCO actions. The results of this data gathering effort can be seen in the Question 5 response.

Table 3.1: STP PSA/IPE Frontline and Support Systems		
•**Frontline**	**Trains**	**Designator**
— Reactor Containment Fan Coolers	A,B,C	HC
— Containment Isolation System	NA	*
— Emerger y Core Cooling System		SI
• Refueling Water Storage Tank and Suction Lines	A,B,C	
• Containment Pump Suction Lines	A,B,C	
• High Head Safety Injection	A,B,C	
• Low Head Safety Injection	A,B,C	
— Auxiliary Feedwater System and Secondary Steam Relief	A,B,C,D	AF
— Containment Spray System	A,B,C	CS
— Primary Pressure Relief	NA	*
— Main Steam Isolation Systems	A,B,C,D	MS
— Chemical and Volume Control System	A,B	CV
— Residual Heat Removal System	A,B,C	RH
•**Support Systems**		
— Essential Chilled Water System	A,B,C	CH
— Electric Power System (including diesel generators)		
• 4,160V BUS	A,B,C	PK
• DC BUS	A,B,C,D	DJ
• Vital AC Channel	I,II,III,IV	VA
• Emergency Diesel Generators	A,B,C	DG
— Component Cooling Water System	A,B,C	CC
— Essential Cooling Water System	A,B,C	EW
— Reactor Trip System	NA	RS
— Solid State Protection System	R,S	SP
— Engineered Safety Features Actuation System		SF
• ESAF Train	A,B,C	
• ESF Load Sequencer	A,B,C	
— Qualified Display Processing System	A,B,C,D	AM3
— Electrical Auxiliary Building HVAC System	A,B,C	HC
— Control Room HVAC System	A,B,C	*
— Instrument Air	NA	IA

* Maintenance scheduled for outages or refuelings.

Table 3.2 : Twelve Week Work Control Schedule*

Week	1	2	3	4	5	6
Train	A	B	C	D	A	B
System	EW CC SI CS RH PK DG CH HC	AM3 SF SP DJ VA IA	AF MS	CC AM3 SP DJ VA CH	AF MS CW IA CV	EW CC SI CS RH PK DG CH IA HC
Week	7	8	9	10	11	12
Train	C	D	A	B	C	D
System	AM3 SP RS DJ VA IA SF	AF MS IA	AM3 SP SF IA CV DJ VA	AF MS SM IA	EW CC SI CS RH PK DG CH HC	N/A

* In general the work activities scheduled during train outages are corrective maintenance (non-LCO related), preventative maintenance, and operability testing via surveillance activities.

4. Provide a detailed discussion of the bases and supporting reasoning for changes to the STP TS for the non-core cooling systems (systems that were not needed to prevent a core damage event).

RESPONSE:

The bases for the requested changes for systems not needed to prevent a core damage event are provided in the "Risk-Based Evaluation of STPEGS Technical Specifications" provided by ST-HL-AE-3283 dated February 1, 1990. These systems are Containment Ventilation and Containment Isolation. These discussions are found in Sections 4.10 and 4.13 respectively of the Study. It is stated on page 2-5 of the submittal that the figure of merit for this Study is the estimated change in core damage frequency. The two systems identified above do not effect core damage frequency but do impact overall risk.

In addition, on page 2-4 of the Study it is observed that STP has more ESF equipment under the scope of the technical specifications than does a more typical two train plant. By having more equipment under technical specifications, STPEGS is required to perform more testing and maintenance. On the other hand, the three-safety-train design at STP means that the loss of a single train of ESF equipment leaves STPEGS with two remaining trains, while a two train plant would generally have only one.

This fact plays an important role with respect to accident sequence success criteria. For all other initiators other than LLOCA STP can be brought to a safe, stable condition with only one train of ESF equipment. That is, except for LLOCA, STP has three redundant trains of ESF equipment, any one of which can be used to shut the plant down. Even in the case of the LLOCA, a single train of ESF equipment will be effective in the likely case that it provides flow to an unbroken loop.

The results of the STP PSA, which is emphasized even more by the IPE submittal, indicates that the likelihood of a core damage event at STP is very low (1.7 x 10^{-4}/year in the PSA and 4.4 x 10^{-5}/year in the IPE submittal). Also, it is considered that a comparison of the sensitivity of core damage at STP to the failure or unavailability of equipment and trains with other domestic Westinghouse plants reveals that STP is substantially less sensitive to equipment failures than at other plants due to the three train safety system design. A review of the risk achievement factors summarized in the STP IPE as compared to other Westinghouse plants should confirm this observation to the reviewer.

Therefore, it is considered that the improved safety of the STP design as shown by the low core damage frequency and the reduced sensitivity of core damage to equipment failures, along with the fact that this improvement comes at the expense of increased maintenance requirements compared to other plants, is sufficient justification for technical specification revisions which specifically reflect the STP design. In this case, the systems in question have no direct relation to core damage.

However, it is recognized that the function of the subject systems relates to risk, not to core damage.

At the time the PSA and the Study were submitted, the risk model included only a Level 1 analysis. However, the NRC has indicated that it does not propose to evaluate the proposed changes on the merits of the previous analysis, but using the risk model which was enhanced for the purpose of the IPE submittal. This enhanced risk model includes a Level 2 analysis, i.e., containment performance analysis, which was requested by GL 88-20.

The new rules based modeling in RISKMAN® 3.06 now enables one to quantify the containment ventilation and isolation functions as they relates to specific plant damage states (i.e., we can now determine the failure contribution of containment ventilation and isolation to release category). In this regard a quantitative basis for the evaluation of the risk impact of changes to these technical specifications is now possible. (This same argument is also true for room cooling (e.g., EAB HVAC) since (via rules) a direct correlation between HVAC failures and plant damage states can be obtained.)

Provided that the NRC considers that additional justification is necessary in order to grant changes to these technical specifications over the qualitative arguments contained in the application, it is suggested that further analysis should be performed with the RISKMAN® Level 2 model to support this review. In addition, since the NRC is proposing to use this enhanced model rather than the basis submitted previously, it would be informative to review the impact of the plant specific data and outage times of less than ten days on the systems previously submitted and subsequently withdrawn (ECW, AFW and SDG). If the results are substantially lower, the potential to consider these systems for changes could be reevaluated.

5. Provide STP-specific data on the repair times (including major findings on causes of component failure) used for the safety trains for which AOT changes are requested. This data should cover a minimum of a 144 week period and should include the repair time (portion of the AOT) for which the train was unavailable (i.e., for which the LCO condition was entered).

RESPONSE:

The following systems were proposed for Allowed Outage Time (AOT) technical specification changes (AFW, SDG and ECW have subsequently been withdrawn):

Chemical and Volume Control,
Pressurizer Safety Valves,
Accumulators,
Emergency Core Cooling,
Residual Heat Removal,
Containment Spray,
Containment Spray Additive (Revised by plant design change),
Reactor Containment Fan Cooler,
Containment Isolation,
Steam Generator Safety Relief Valves,
Auxiliary Feedwater,
Component Cooling Water,
Essential Cooling Water,
Control Room HVAC,
Essential Chilled Water,
Diesel Generators, and
DC Electrical Sources.

The above question is answered by calculating the average out of service (OOS) duration for each system. At the South Texas Project (STP), an accountability process exists that will provide the OOS durations. This method entails using the Operability Tracking Log (OTL), which contains the equipment information, LCO conditions, and the time a system is removed from service and then returned to service. From these logs it is possible to determine a reasonable train outage time for each of the systems. This information provides a clear picture of plant specific system down times, which bounds the requested component repair time. For example, a pump might take 20 minutes to repair and then another 2 hours to reconnect back to the system (i.e, return to service). So, instead of providing an unavailability due to repair times, a more meaningful unavailability is provided for the system out of service time

With respect to repair times, several work packages were pulled and evaluated for repair times. These packages did not detail any information on a time frame for repairing a component. Instead, the amount of man-hours to process the work package, including actual repair time, was given for each individual who worked on the component. These man-hours take into account items such as obtaining parts from the warehouse, obtaining permission from the control room and various other administrative items not related to actual hands-on repair times. The work request package is the most detailed document on repairing components; however, plant specific data on repair times for individual components are not readily retrievable from the packages due to the process used to account for expended person-hours. Further work and possibly detailed work package evaluations would be required to provide, at best, an estimate of repair times.

With this in mind, HL&P considers that the OOS time duration more accurately reflects equipment outage times since return to service occurs usually just prior to operability declarations (i.e., equipment is made functional with return to service and then operable with surveillance testing).

Evaluation for several LCO conditions could not be performed. These include the following systems: Pressurizer Safety Valves, Accumulators, Containment Spray Additive, Containment Isolation, Steam Generator Safety Relief Valves and Control Room HVAC. The predominant reason for non-evaluation entails that scheduled maintenance OOSs were performed under different technical specifications. Also, some of these have preventative maintenance and surveillance activities which are performed only during refueling or shutdown periods.

The following steps characterize the mean time of unavailability for each system.

1. Find the Data
2. Evaluate the Data
3. Determine the Lognormal Parameters
4. Plot the Data

1. Find the Data

The data source for the OOSs was found in the OTLs for each unit from November 1989 to August 1992. This data reflects approximately 144 weeks for each unit and totals 288 weeks of reactor data. The use of more than 5 years of reactor data will generate a distribution for the requested systems.

2. Evaluate the Data

There are several assumptions made in evaluating the plant specific data used in answering the request for additional information.

A. That more than 5 years of reactor data is adequate to develop a plant specific distribution.

B. The OTL is assumed to reflect actual system outage duration. This is a valid assumption since the OTLs keep track of system outages which impact LCOs at STP.

C. Since the PSA is an "at power" model, all data collected during a non-mode 1 is screened out.

D. This data reflects only outages that are consistent with the technical specifications proposed for change.

3. Determine Lognormal Parameters.

This part of the process entails evaluating the data to determine the mean OOS duration for each of the safety systems. Industry and RISKMAN standards require an evaluation of the data using the lognormal distribution. This is reasonable since the distribution of an infinite number of data points would be expected to approximate the lognormal distribution.

4. Plot the Data

The data is plotted as the relative unavailability of each system train verses the OOS time in hours. Figures 1 through 11 display the data in a graphical form. Table 1 presents the lognormal mean, in hours, of train-wise outages, along with the 5^{th}, 50^{th} and 95^{th} percentile of the lognormal distribution.

Question #5 also asks for the 'major findings on cause of component failures.' Table 2 displays the results of searching the NPRDS database for the South Texas Project.

Finally, Table 3 provides a summary of the trainwise unavailabilities on an annual basis. These "experienced" unavailabilities are substantially less than those assumed in the PSA.

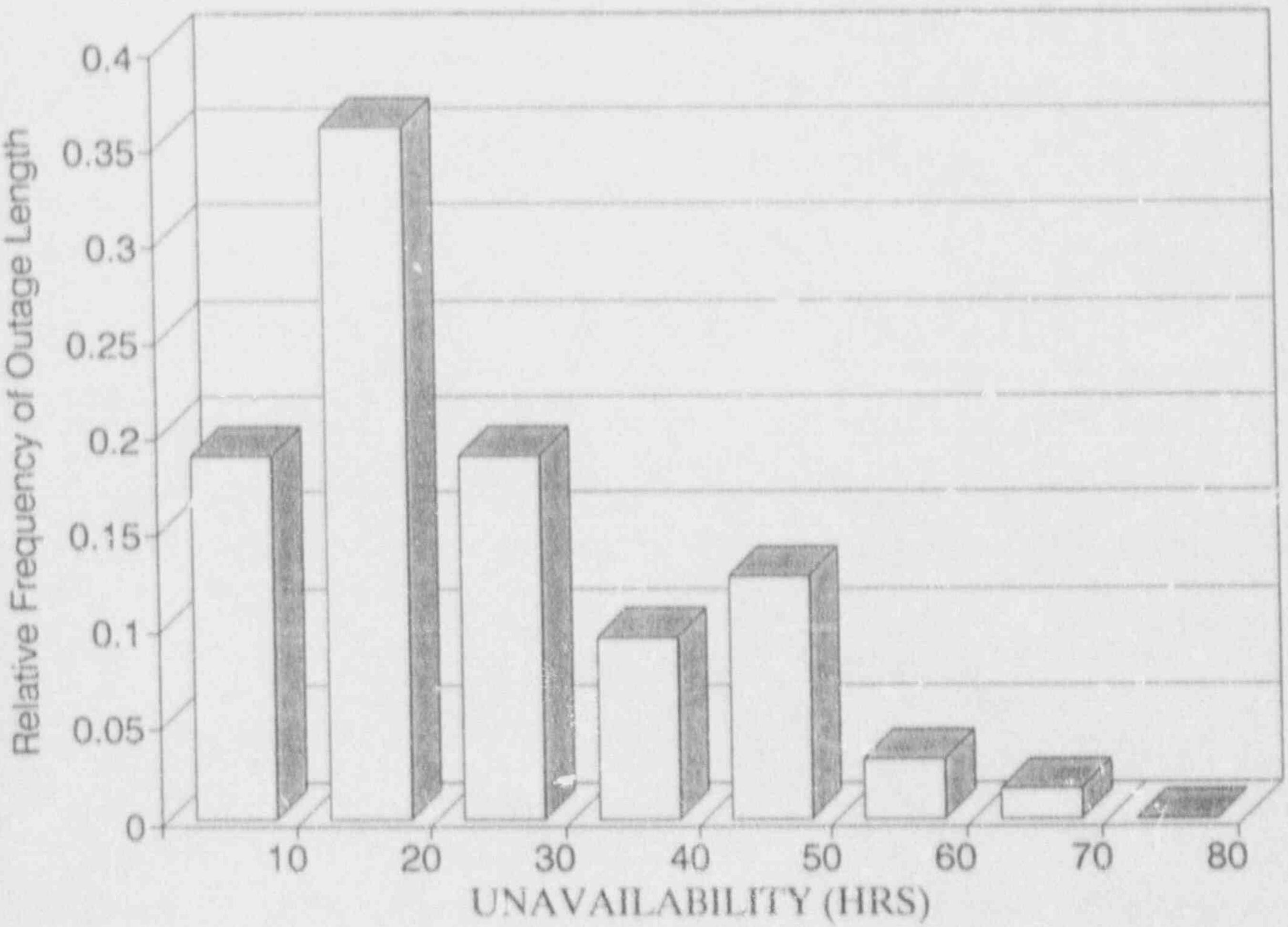
Table 1: Parameters (hrs) from the Lognormal distribution				
	Mean	5^{th}	50^{th}	95^{th}
Auxiliary Feedwater	22.8	6.9	18.8	50.2
Component Cooling Water	26.3	10.3	23.1	50.9
Containment Spray	31.7	12.2	27.8	62.1
Chemical and Volume Control	30.9	15.1	28.5	53.1
DC Electric Sources	11.6	1.7	7.8	33.0
Diesel Generators	30.6	8.8	24.9	68.8
Emergency Core Cooling	23.7	8.1	20.2	49.0
Essential Cooling Water	37.2	15.5	33.2	69.9
Essential Chilled Water	29.6	7.6	23.4	69.5
Reactor Containment Fan	21.1	6.6	17.6	45.6
Residual Heat Removal	26.1	9.4	22.5	52.7

Table 2: Major findings on causes of component failures		
Auxiliary Feedwater	Normal Wear Cyclic Fatigue Manufacture defect Unknown Other	6 1 1 1 1
Chemical and Volume Control	Normal Wear	1
Residual Heat Removal	Normal Wear	1
Component Cooling Water	Normal Wear Unknown Other	7 1 2
Containment Spray	Normal Wear	1
Diesel Generator	Normal Wear Fatigue Manufacture defect Maintenance Induced Unknown Other	12 2 1 1 7 1
DC Electric Sources	Normal Wear	2
Essential Cooling Water	Normal Wear	1
Reactor Containment Fan Coolers	Normal Wear Manufacture Defect Unknown Other	3 2 1 2
Emergency Core Cooling	Normal Wear Maintenance Induced	1 1
Main Steam	Normal Wear Cyclic Fatigue Maintenance Induced Unknown Other	1 1 1 1 2
Chilled Water	None*	
Control Room HVAC	None*	
Electrical Aux HVAC	None*	

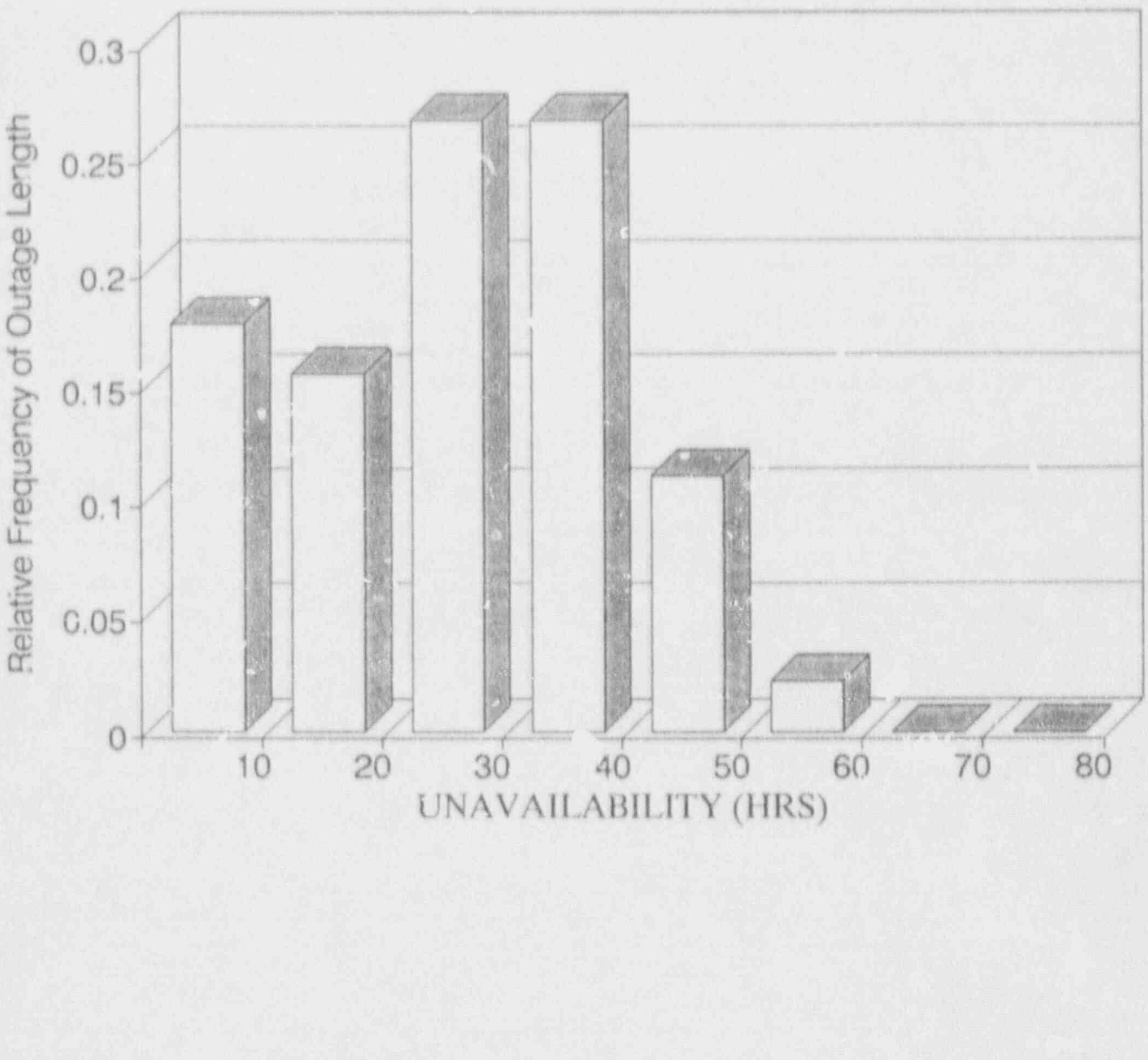
* No failures found in NPRDS using search criteria.

Table 3: Train Unavailability on an Annual Basis	
System	Annual Unavailability per Train
Auxiliary Feedwater	1.010e-02
Component Cooling Water	8.172e-03
Containment Spray	3.285e-03
Chemical and Volume Control	3.421e-03
DC Electric Sources	4.798e-04
Diesel Generator	2.200e-02
Emergency Core Cooling	7.038e-03
Essential Chilled Water	1.760e-02
Essential Cooling Water	1.440e-02
Reactor Containment Fan Coolers	2.921e-03
Residual Heat Removal	2.889e-03

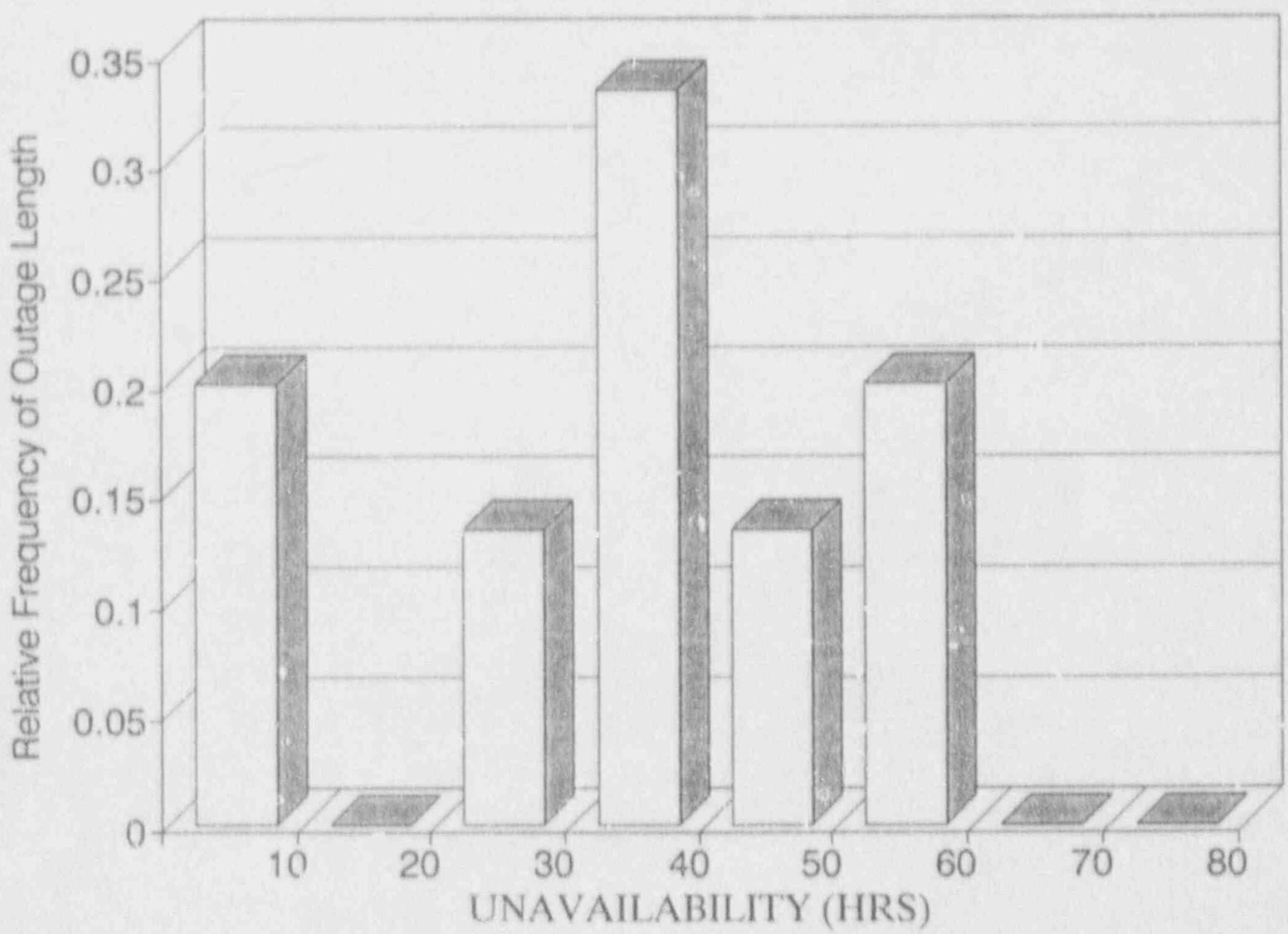
AUXILIARY FEEDWATER SYSTEM
UNAVAILABILITY FROM 11-89 TO 8-92



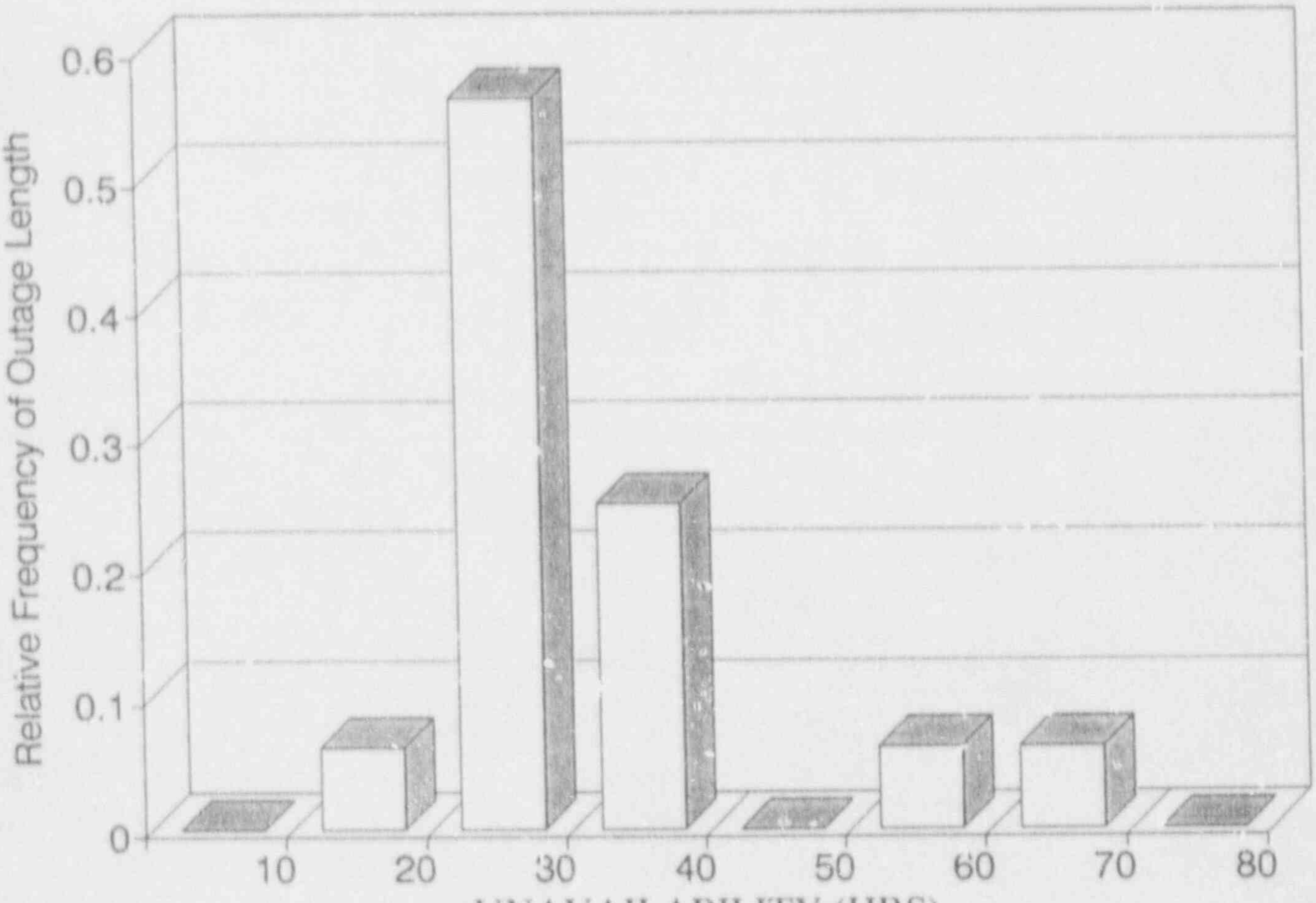
COMPONENT COOLING WATER SYSTEM
UNAVAILABILITY FROM 11-89 TO 8-92



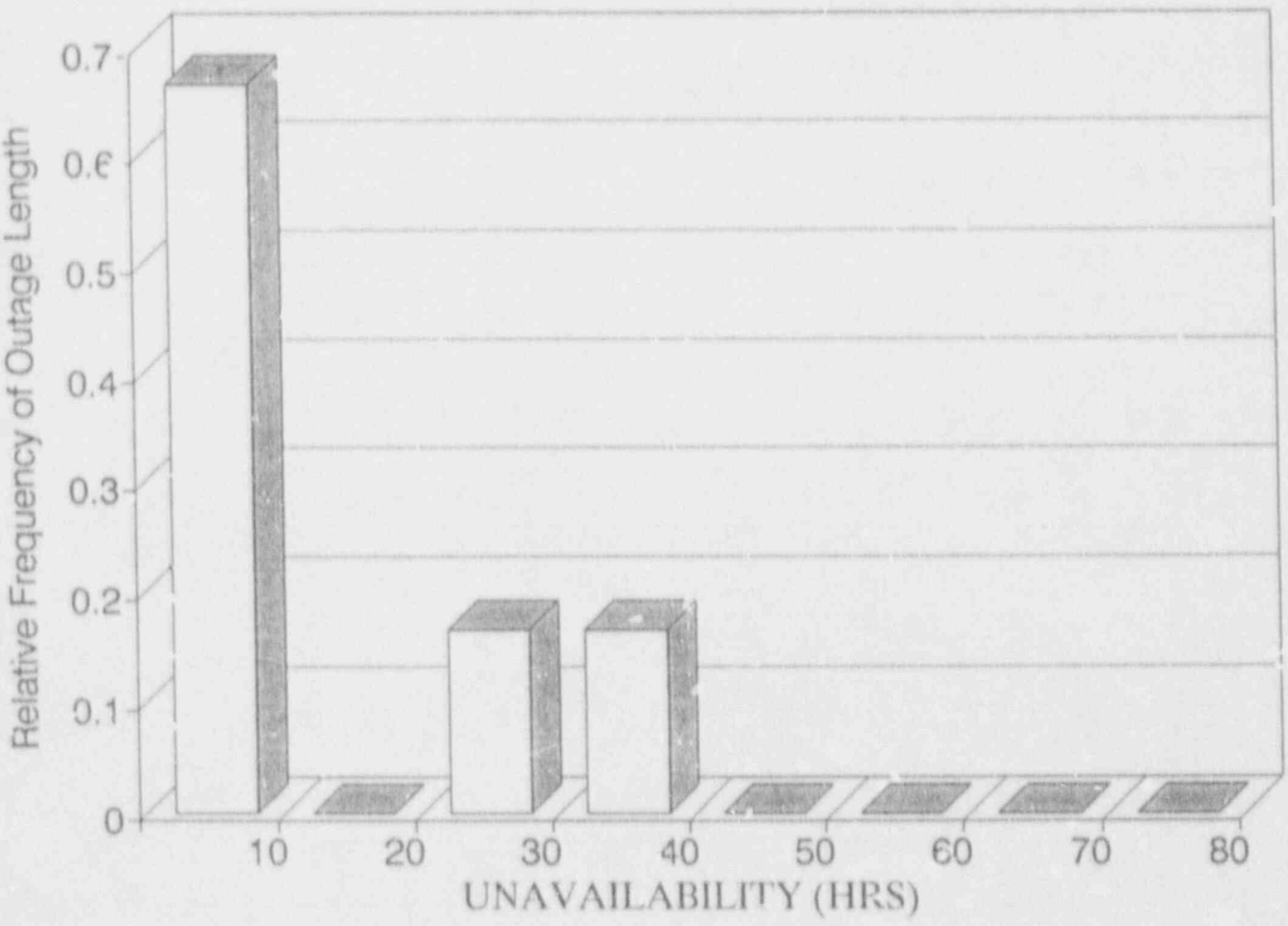
CONTAINMENT SPRAY SYSTEM UNAVAILABILITY FROM 11-89 TO 8-92



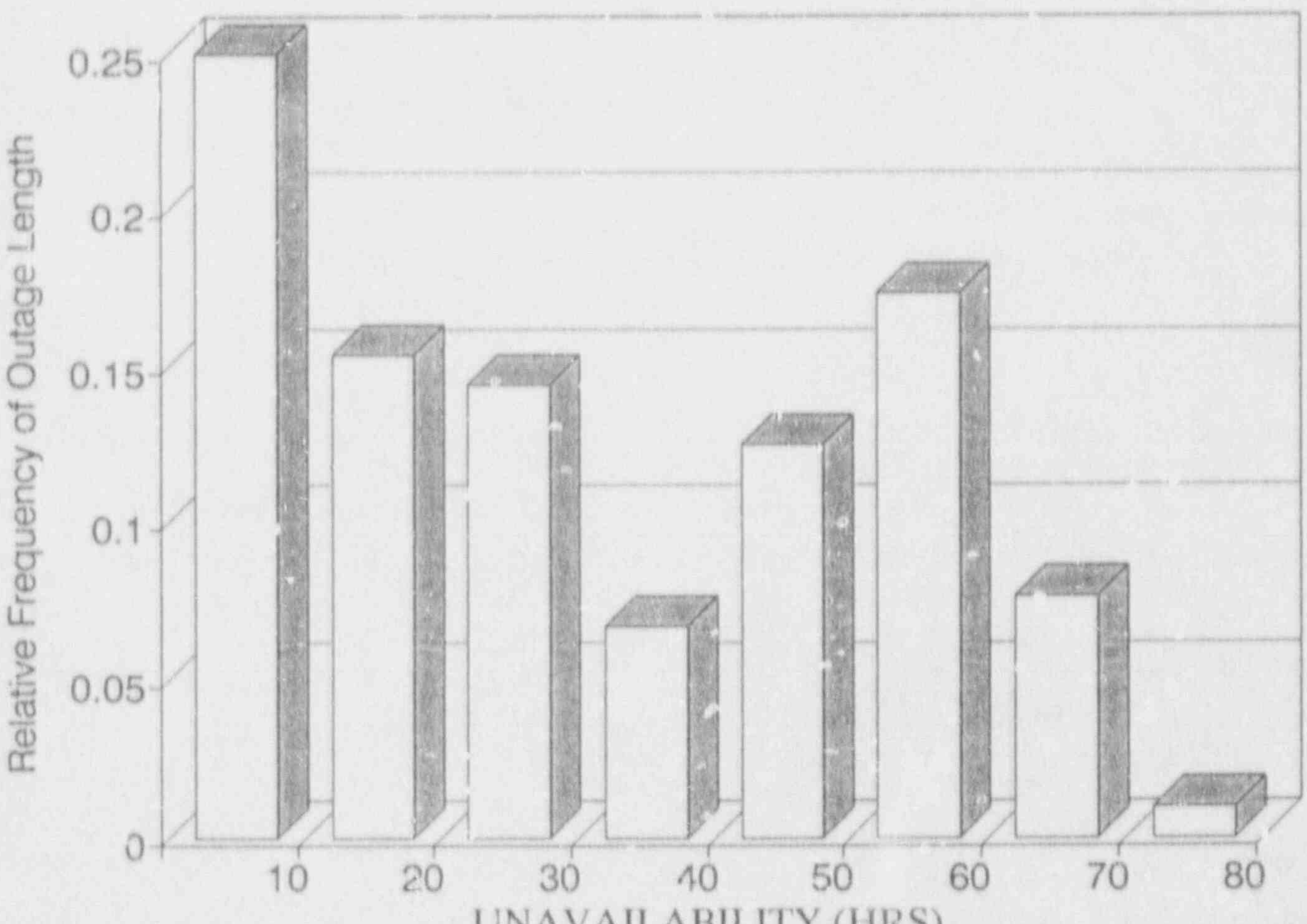
CHEMICAL AND VOLUME CONTROL SYSTEM UNAVAILABILITY FROM 11-89 TO 8-92



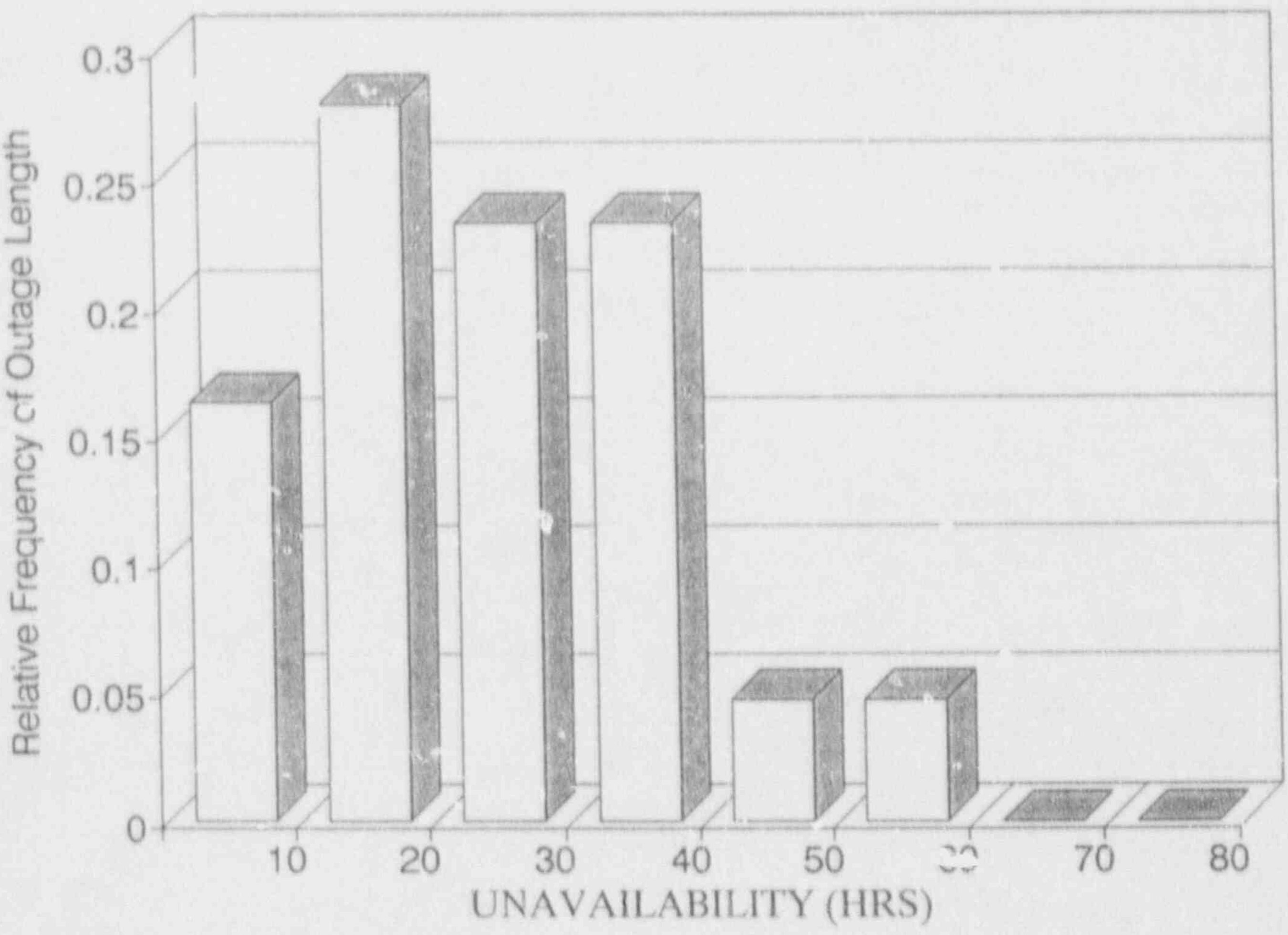
DC ELECTRIC POWER SYSTEM
UNAVAILABILITY FROM 11-89 TO 8-92



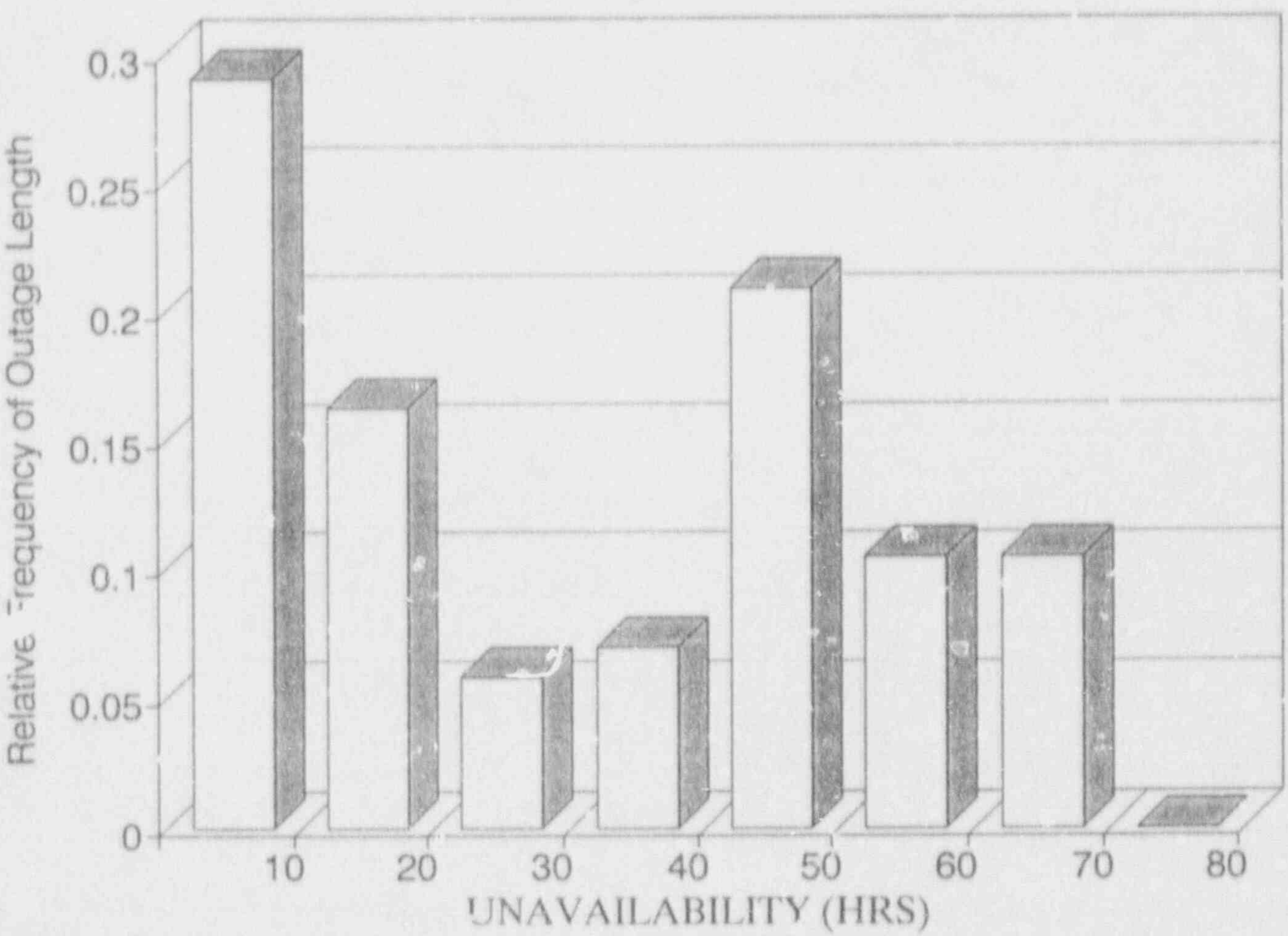
DIESEL GENERATORS
UNAVAILABILITY FROM 11-89 TO 8-92



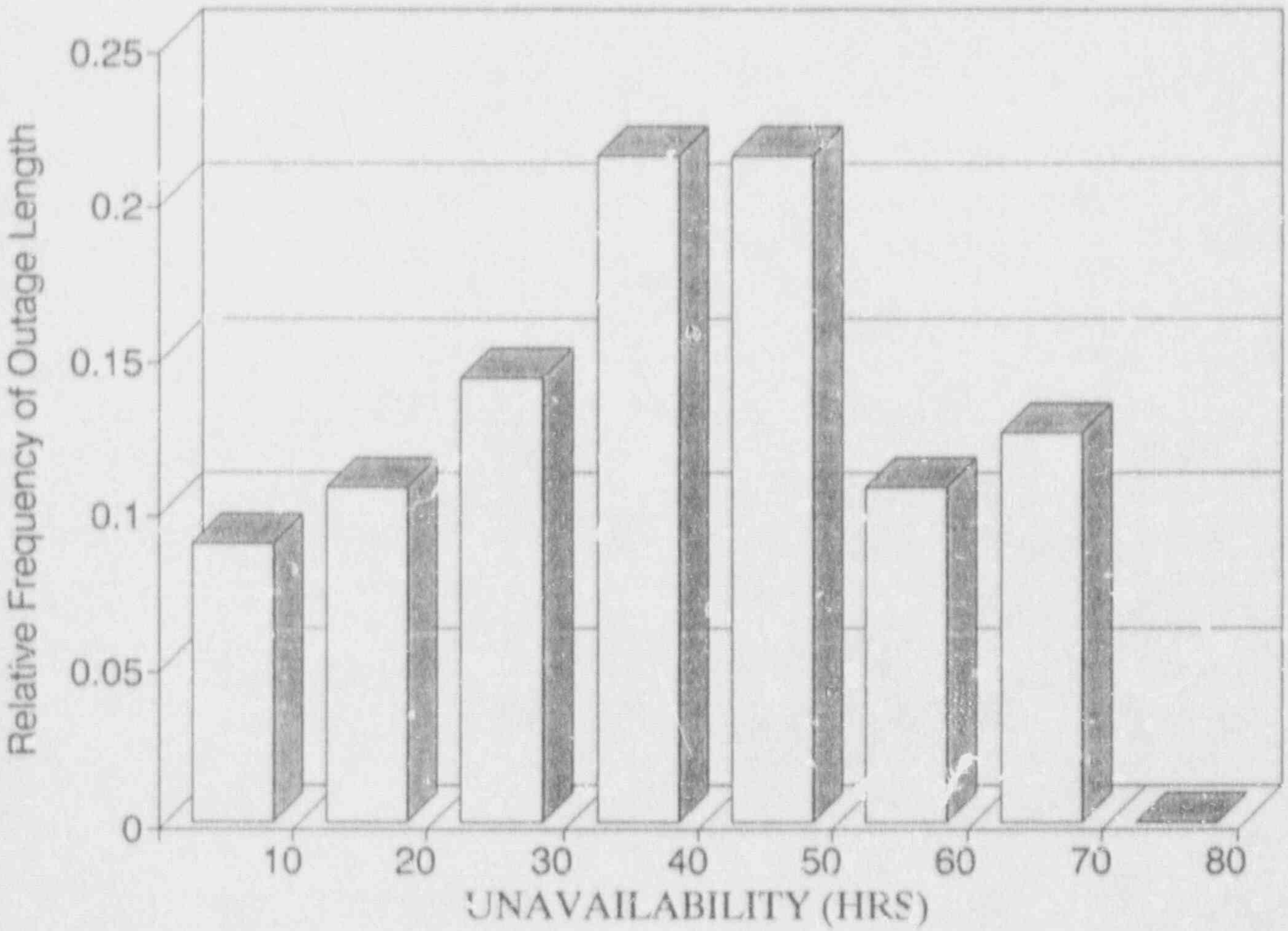
EMERGENCY CORE COOLING SYSTEM UNAVAILABILITY FROM 11-89 TO 8-92



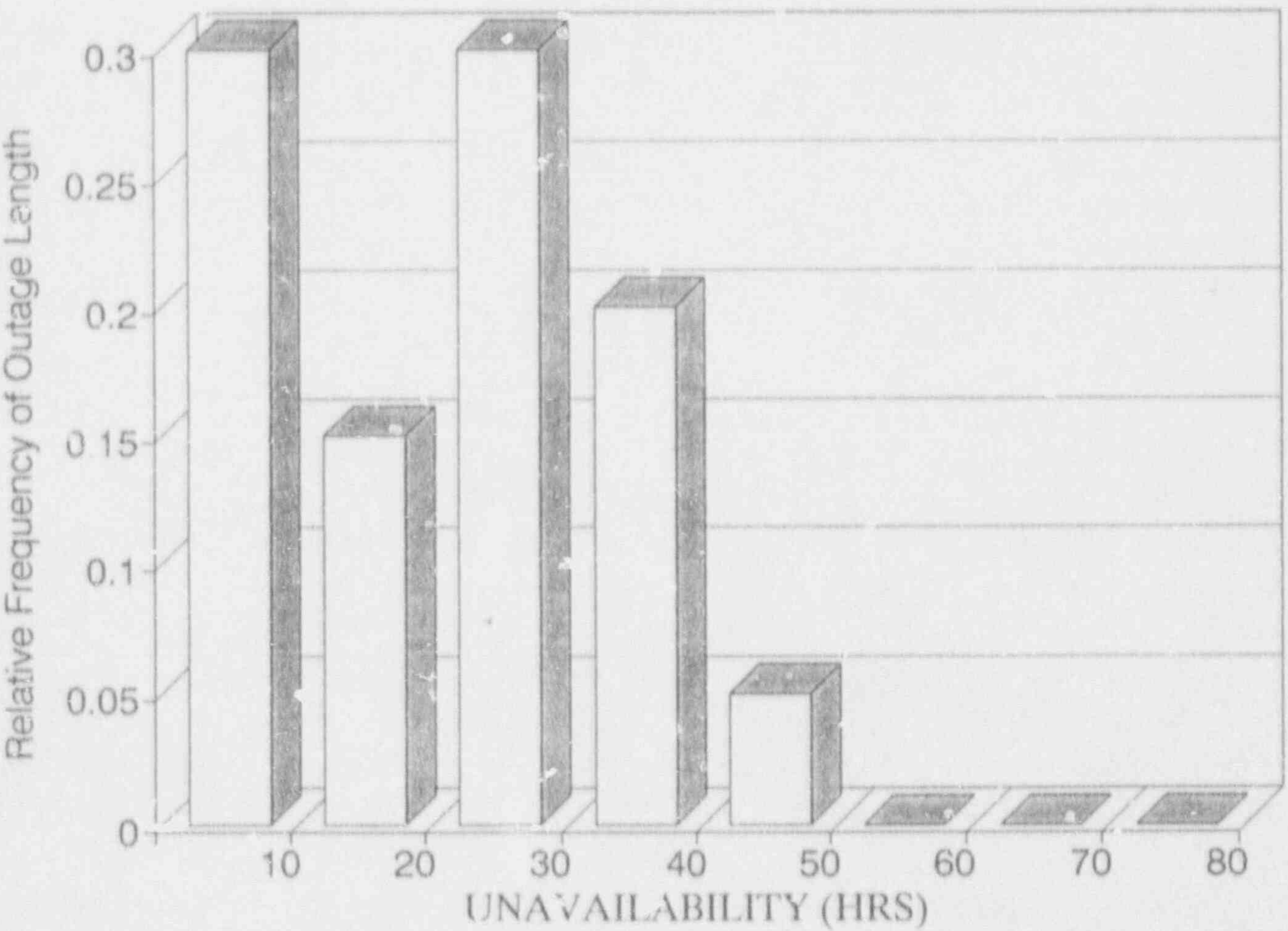
**ESSENTIAL CHILLED WATER SYSTEM
UNAVAILABILITY FROM 11-89 TO 8-92**



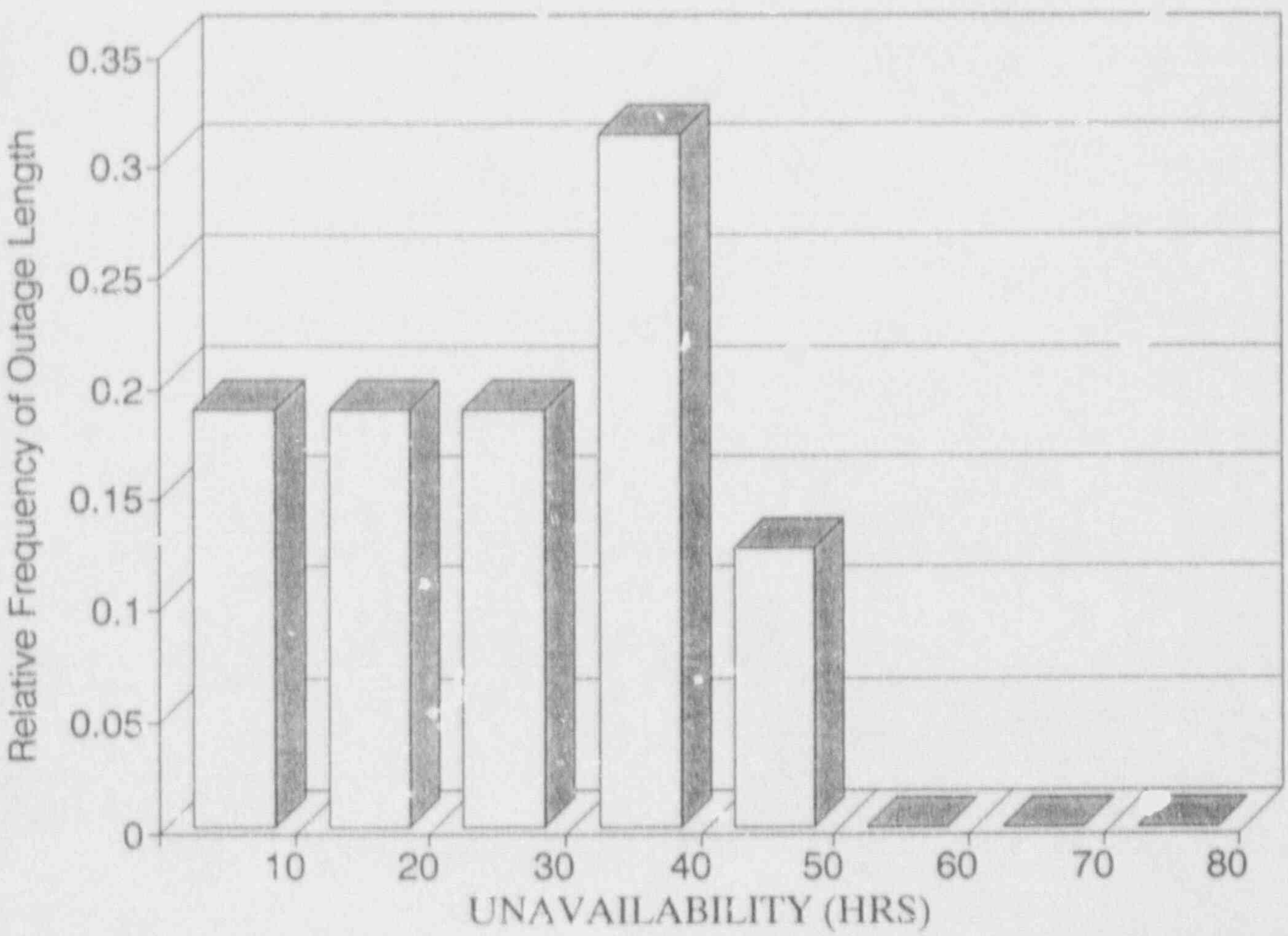
ESSENTIAL COOLING WATER SYSTEM
UNAVAILABILITY FROM 11-89 TO 8-92



REACTOR CONTAINMENT FAN COOLERS SYSTEM UNAVAILABILITY FROM 11-89 TO 8-92



RESIDUAL HEAT REMOVAL SYSTEM
UNAVAILABILITY FROM 11-89 TO 8-92



6. Were safety system train configurations changed as part of the development of the proposed changes to the current TS? If so, provide a list of those reconfigurations along with the supporting reasoning.

RESPONSE:

Safety system train configurations were not changed as part of the development of the proposed changes to the current Technical Specifications. In general, any system or train manipulations for the purposes of performing surveillance testing activities or maintenance are reflected in the system specific alignments calculated in the system analyses. These alignments determine the time duration that a system/train is in a "testing" or "maintenance" alignment with any associated unavailability transferred to the appropriate top event split fractions.

7. Provide a summary of the results of studies performed to analyze the risks associated with power reductions or other routine plant transients.

RESPONSE:

In April of this year (1992), the Plant Analysis Division performed a review of plant trips at STP which have occurred since plant startup, including both STP units. This review was used to update the initiating event frequencies for the IPE submittal. The review included all plant trips, prior to and subsequent to commercial operation.

A total of 41 plant trips were reviewed. Of these, three were found to be related to power level changes. The attached "Table 4, STPEGS Plant Trip Review, Trip Events Associated With Power Ascension," provides a brief description of the trips identified. Although one of these trips occurred at 98% power (ascension from lower power), it was also concluded that small changes in power above 90% were likely not susceptible to increase risk.

Other impacts include

- Effects on power distribution and control rod usage
    - During power reductions the existing unequal quadrant core burnups cause quadrant power and xenon distributions to change by different amounts. This phenomenon can result in quadrant tilt ratios exceeding Technical Specification limits.
    - Axial Flux Difference (ΔI) control during power reductions results in increased rod usage which could result in an increased potential for misaligned and/or dropped rods.
- Fuel Integrity and Dose Equivalent Iodine
    - Power cycling from compliance with LCO action statements have adverse effects on fuel integrity and the resultant RCS radiochemistry levels.
- Increased demands on operators, engineering, and maintenance staff due to implementation of LCO actions (e.g., nuclear instrumentation Power Range calibrations and Hi Flux Rate Reactor Trip Setpoint for power ascension activities).
- Increased demands on equipment (e.g., actuation of MSR steam vent valves to the Feedwater Heater position and starting additional Feedwater pumps for power ascension activities).
- Mobilization of Reactor Plant Operator and Auxiliary Operator staff for local equipment monitoring and actions.

TABLE 4

STPEGS PLANT TRIP REVIEW

TRIP EVENTS ASSOCIATED WITH POWER ASCENSION

PLANT TRIP	POWER LEVEL	CAUSE OF TRIP	REMARKS
1-011	76%	Hardware Failure. EHC line failure.	Although these lines have since been modified, the EHC controller will always be challenged as a direct result of power level changes. Therefore, this item counted as a power ascension trip.
1-012	98%	Human Error. 2/4 OTDT trip. One channel tripped for calibration, PZR pressure control response brought in the other channel.	This human error event is a direct result of a calibration performed due to transient maneuvering. Thus, this item is counted as a power ascension trip.
1-017	40%	Hardware Failure. Dropped control rod with subsequent reactor trip on hi neg. flux rate.	Startup procedure (ZG-00(5) was in progress at the time of the trip. The dropped rod was due to failure of diodes which are demanded during rod control movements. This trip is counted since additional power reductions will increase the number of demands on these mechanisms.