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APPENDIX B

COMMONWEALTH EDISON COMPANY

OFFSITE DOSE CALCULATION SYSTEM

A Meteorological Monitoring, Offsite Dose Calculation Program for Emergency Preparedness at Operating Nuclear Power Plants

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Comparison between NRC Zion Order Dated 2/29/80
and the CECo. Offsite Dose Calculation System

Zion Order	ODCS (or comment)
Appendix A, Item F.2	
a. Meet criteria of Annex 2 pending necessary equipment installation and other work	a. See following comments on Annex 2.
b. During interim period provide plant with real-time forecast provided by consultant.	b. A real-time forecast will be provided by Murray and Trettel, Inc., certified consulting meteorologists
Annex 2	
1.a.	CECo. will meet this criterion ("Will meet"). The schedule for meeting this or other sections of the order is given in section VII of the ODCS report.
1.b.	Will Meet
1.c.(1)(a)	Will Meet
1.c.(1)(b)	Will Meet
1.c.(1)(c)	Will Meet
1.c.(1)(d)	Will Meet
1.c.(1)(e)	Will Meet except for the dew-point detector (a sheltered Foxboro dew cell) which is at 2 meters not 10.
1.c.(1)(f)	Will Meet
1.c.(1)(g)	Will Meet
1.c.(2)	Section 2.3.3. of NUREG-75/087 refers to RG 1.23, a document whose guidance on equipment CECo. meets except for the dew cell.

Zion Order	ODCS (or comment)
1.c.(3)	CECo has had a formal QA program for its meteorological facilities since 1976. (See ODCS report Section V and Appendix G). However, the requirements of Revision 1, Section 17.2 of NUREG-75/087 cannot be applied in their entirety to the meteorological program, a professional consulting service. Nonetheless the CECo QA program will be upgraded significantly as is indicated by comparing Rev. 0 of the QA Articles to Rev. 1 in Appendix G. This upgraded program will be implemented by 1/1/81.
1.c.(4)	A redundant power source will be provided either by connection to electrical power system or installation of an electrical generator equipped with an automatic start mechanism at the tower.
2.a.	Will Meet. See ODCS report Section III. 2 and Appendices C and D.
2.b.	Will Meet
2.c.(1)	Will Meet
2.c.(2)	Will Meet
2.c.(3)	Will Meet
2.c.(4)	Will Meet the equipment requirements except for backup dewpoint sensor at Zion.
2.c.(5)	See CECo comment on Annex Section 1.c.(3) above.
2.c.(6)	Will Meet

Zion Order	ODCS (or comment)
3.a.	Will Meet
3.b.	Will Meet
3.c.(1)	Will Meet. See ODCS report Section III.1 and Appendices A and B.
3.c.(1)(a)	Will Meet
3.c.(1)(b)	Will Meet
3.c.(1)(c)	The CECo - NRC data transmission link is still under development. See ODCS report Section III.5.
3.c.(2)	Will Meet. See ODCS report Section III.3. The CECo forecast will be for 36 hours at hourly intervals.
3.c.(3)	Will Meet. See ODCS report Section IV and Appendices E and F.
4.a-c(5)	The NRC's Nuclear Data Link remote interrogation system is still being developed by the NRC. No commitment on the NDL can be made until this system's characteristics are more fully described and evaluated.

Commonwealth Edison Company
Offsite Dose Calculation System

I. Introduction

NUREG-0654 "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" and the NRC order for Zion Station dated 2/29/80 describe meteorological criteria for emergency preparedness at operating nuclear power plants and Zion in particular. The position of the NRC is that all operating plants shall have an adequate operational meteorological measurements program to produce real-time and record historical local meteorological data. Highlights of these criteria are:

(1) There shall be a primary meteorological measurements program and a viable backup system and/or procedures to obtain real-time local meteorological data.

(2) There shall be a QA program consistent with applicable provisions of Appendix B to 10 CFR 50; the acceptance criteria of Revision 1, Section 17.2 of NUREG-75/087 apply.

(3) The metro tower(s) shall be connected to a power system which is supplied from redundant power sources.

(4) There shall be two classes of atmospheric dispersion models:

Class A: A model which can produce initial transport and diffusion estimates within 15 minutes following classification of an incident.

Class B: A model which can produce refined estimates for the duration of the release. It shall also include forecasts of changing metro conditions.

(5) The models shall incorporate these features: weather forecasts, (for the Class B model only), local meteorological anomalies (such as lake effects at Zion), routine meteorological data transmission to the NRC, and simultaneous remote interrogation by the licensee, the emergency response organization and the NRC.

This report describes the Commonwealth Edison (CECo) Offsite Dose Calculation System (ODCS), a computer-based method for estimating the environmental impact of unplanned airborne releases of radioactivity from nuclear stations. The ODCS is designed to meet the meteorological criteria of NUREG-0654 and the NRC order for Zion Station.

II. Objectives of the ODCS

The objectives of the Commonwealth Edison Offsite Dose Calculation System are:

(1) Meet the meteorological criteria of NUREG-0654 and the Zion order.

(2) Provide, where possible, redundant independent pathways of data transmission and redundant data processing computers for use in an emergency situation.

(3) Provide quick and reasonably accurate estimates of radiation dose to persons living offsite, including preparation of procedures and training of users required to accomplish this assessment.

(4) Provide a method for the meteorological contractor to secure meteorological data for assessment of routine releases and to detect equipment failure quickly.

III. Description of the O.D.C.S.

1. System Design and Atmospheric Dispersion Models

Design

On a routine basis each nuclear station meteorological tower will be interrogated many times daily by the meteorological contractor to secure the information necessary for preparation of meteorological operating reports and to detect system failures.

Every hour, and more frequently during an accident, a corporate (in Chicago) SYFA computer will poll each meteorological facility to prepare the corporate data file and to check the system in order to maintain the ODCS in a readiness posture. Corporate IBM computers will then store the data for an extended period of time and process the data when refined estimates of dose are needed.

At each nuclear station, two computers with different functional requirements will process the meteorological information. The plant process computer will produce initial transport and diffusion estimates within 15 minutes following classification of an incident. The plant SYFA computer will produce refined estimates of dose in two ways: (1) as a terminal entry system to the corporate IBM or (2) by itself when the data link between the plant's Technical Support Center (TSC) and the corporate IBM is lost.

During an accident these four computer systems (plant process, plant SYFA, corporate SYFA, corporate IBM) will provide the various users with timely information required to make decisions. Emergency actions will be performed in the following sequence:

first - time frame: initial one-half hour or so post-accident - the control room operator will rely on wind speed and direction and effluent release rate information provided by the plant process computer and these data converted into requisite Emergency Action Levels (EAL) by the Class A computer model.

second - 1/2 hour to few hours - the plant will rely on the station-designated ODCS user to analyze the off-site consequences using the corporate IBM computer (Class B model) or plant SYFA (Class B' model).

third - few hours to duration of accident - a corporate environmental group will perform refined estimates of the offsite consequences for the duration of the emergency period using effluent information provided by plant personnel and the corporate IBM Class B model. This corporate group has been formed to support all nuclear stations and will perform its work in Chicago in lieu of having to relocate to each nearsite Emergency Operations Facility (EOF). A data link between the corporate facility and each EOF will be provided.

Figure 1 shows the ODCS data processing centers and the multi-tiered lines of communication for transmitting meteorological information among the centers. The control room operator will be provided ODCS information from the plant process computer which will be linked directly to the meteorological tower. The operator will have immediately available on command meteorological, noble gas effluent, emergency action level, and offsite dose consequence information through the Class A computer model.

Table 1 provides a summary of CECo's planned Offsite Dose Calculation system.

The backup meteorological measurements program, forecasts of changing meteorological conditions, the NRC data link for meteorological information, and if installed the Zion Station Atmospheric Release Advisory Capability (ARAC) are described in subsequent sections.

Models

The Class A model will activate the necessary EAL alarms for site emergency: 2-minute average noble gas release rate having projected offsite dose rate of 500 mR/hr and 30-minute average noble gas release rate having projected offsite dose rate of 50 mR/hr, using worst case meteorology, and for general emergency: 2-minute average noble gas release rate having projected offsite dose rate of 1000 mR/hr using 15-minute average actual meteorology. For additional information on this model see Appendix A.

The ODCS operator in the TSC will have access to the plant process computer, the plant SYFA computer, and the corporate IBM computer. As a result, the TSC operator can produce estimates of the offsite consequences with any or all of the computer program models (Class A, B or B'). Site meterological data will be made available to the TSC through a number of paths: directly via the process computer, directly via the B Microtel to the plant SYFA, or indirectly to the plant SYFA via either the telephone to the Microtels or microwave link to the corporate IBM computer. In addition, meteorological information from other similarly equipped CECo meteorological towers (there are a total of six) may be interrogated via the microwave link between the plant SYFA and the corporate IBM computer. The IBM computer will store minute values of each meteorological parameter, except at this time dew point temperature at Dresden and Zion, for 10 days and hourly averages for 60 days. Dew point temperatures at these two stations will be stored for these same time periods if they are available via the microwave-Microtel link shown in Figure 1. (At Dresden and Zion dew point will be available through Microtel A but not through Microtel B due to limited rack space. However, at this time the location of the microwave tie-line has not been established.)

The Class B model will incorporate techniques for making refined estimates of offsite radiation doses. The corporate IBM computer will have access to all metro facilities to permit study of regional historical, current period, and forecasted weather conditions, have an edit feature to permit the user to correct possible inaccurate metro information due to malfunctioning equipment, have dose models for calculating the offsite individual's whole body dose, the population whole body dose, the individual's skin dose, and the inhalation dose to 7 organs of the adult and child from 73 different non-noble gas nuclides. The B model is described in Section 9.0 of the Offsite Dose Calculation Manual being developed for use in conjunction with the 10 CFR 50 Appendix I Technical Specifications. The current draft of this section is attached as Appendix B. At this time it is not known if the SYFA can carry the entire Class B model, hence, the label B'. The B' model has been subdivided into 9 parts ranked in order of importance to the dose assessment operator. If possible, all parts except the population dose estimate will be placed in the SYFA.

All three computer models used by CECo are based on atmospheric transport models and data processing techniques described in TID-24190 "Meteorology and Atomic Energy 1968", NUREG/CR-0936 "Recommendations for Meteorological Measurement Programs and Atmospheric Diffusion Prediction Methods for Use at Coastal Nuclear Reactor Sites", and Nuclear Regulatory Commission Regulatory Guide 1.109 "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I".

At Zion Station, a dispersion model which accounts for the lake breeze effect off Lake Michigan has been incorporated in one form or another into each model developed from NUREG/CR-0936 class. This lake effect model is described in greater detail in section II.4.

Plume meander or absence thereof as estimated from the measurement of sigma-subtheta has been incorporated into the plume centerline dispersion model which heretofore was based only on a measurement of differential temperature on the tower.

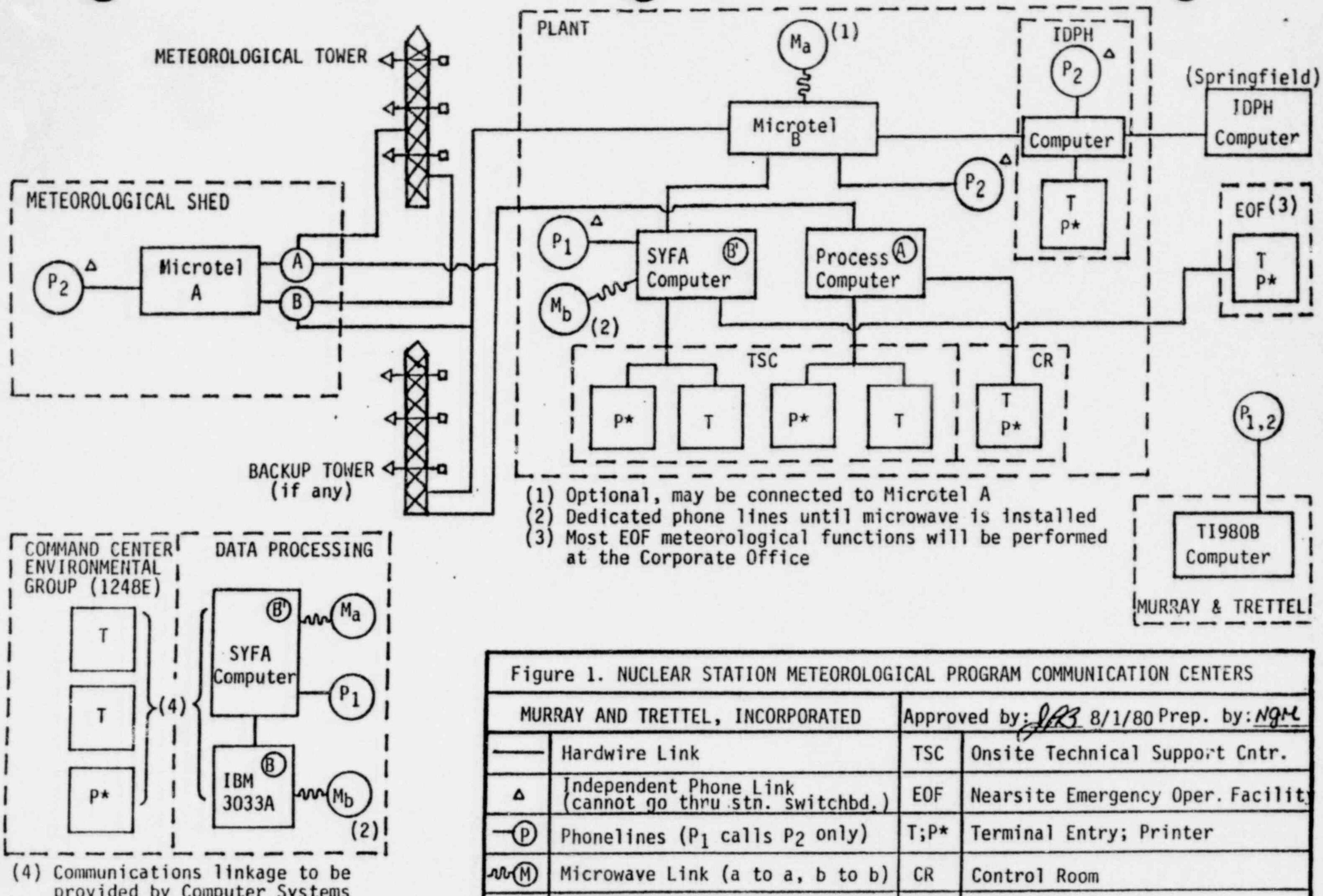


Figure 1. NUCLEAR STATION METEOROLOGICAL PROGRAM COMMUNICATION CENTERS			
MURRAY AND TRETTEL, INCORPORATED		Approved by: JRB 8/1/80 Prep. by: NGM	
———	Hardwire Link	TSC	Onsite Technical Support Cntr.
Δ	Independent Phone Link (cannot go thru stn. switchbd.)	EOF	Nearsite Emergency Oper. Facility
—(P)	Phonelines (P_1 calls P_2 only)	T;P*	Terminal Entry; Printer
∿(M)	Microwave Link (a to a, b to b)	CR	Control Room
A,B	Redundant MET tower signal paths	IDPH	Illinois Dept. of Public Health
(A)(B)(B')	Computer Models		

Table 1

A Summary of the

The Offsite Dose Calculation System

Computer	Source of Meteorological Information: Site	Other Sites	Forecast	Radiation Dose Model	User or Data Link
Plant Process	(1) Direct analog signal from tower	-	(1) Manual entry for lake effect parameters (Zion only)	A	.Control room oper. .TSC ODCS oper.
Plant SYFA	(1) Plant Microtel (2) IBM 3033	(1) IBM 3033	(1) M&T on command (2) IBM 3033 on command (3) Manual entry of entire forecast	B	.TSC ODCS oper.
Tower Microtel	(1) Direct analog signal	-	-	-	Backup Data Link for: .Plant SYFA .Corporate SYFA .Metro Contractor
Plant Microtel	(1) Direct analog signal from tower	-	-	-	Data Link for: .Plant SYFA .IDPH .Metro Contractor .Corporate SYFA

Table 1

A Summary of the

The Offsite Dose Calculation System

Computer	Source of Meteorological Information: Site	Other Sites	Forecast	Radiation Dose Model	User or Data Link
Corporate SYFA	(1) Microwave to one Microtel (M) (2) Phone link to plant M (3) Phone link to Tower M	Same as site. (In each case the polling will be done every hour automatically)	(1) M&T polled automatically every 12 hrs. (2) Manual entry of entire forecast	B'	.Corporate ODCS oper. .NRC data link for metro .ARAC data link for Zion
Corporate IBM 3033	(1) Corporate SYFA polled automatically every 24 hrs. (2) Station SYFA via microwave	(1) Corporate SYFA polled automatically every 24 hrs.	(1) Corporate SYFA polled automatically every 24 hrs. (2) On command polling of plant SYFA (3) Manual entry of entire forecast	B	.TSC ODCS oper. .Corporate ODCS oper. .Other Station .TSC ODCS oper. .EOP ODCS oper.

2. Backup Measurement Systems

Section 2 of Annex 2 to the Zion 2/29/80 order and NUREG-0654, Appendix 2, both require a backup metro measurements program consisting of either a "viable backup system and/or procedures." Although the order calls for a system and/or procedures, CECo will implement both. The backup systems consist of the already existing multiple measurement tower-mounted equipment that is being specially isolated to provide completely independent signals from one another. (Tables 2a - 2f). Therefore, loss of any signal due to component failure will not result in the loss of additional signals. This method of signal isolation is superior to the installation of more instrumented towers in several ways.

First, based on more than 50 station years of operation, instrument failures from whatever cause have occurred in the sensor and/or signal conditioners, thereby preventing other unaffected sensors' signals from being processed. The isolated signal processing with independent power supplies and signal paths is designed to prevent this failure mode from occuring. (For a more detailed description of the independent signal pathways see Appendix C).

Second, a disaster of sufficient magnitude to render all measurements on the tower useless, although extremely remote, would in all probability also inflict similar damage to any backup tower nearby.

Third, CECo's existing instrumented towers at six (6) nuclear plant sites located in Northern Illinois provide an uncommonly high-density measurement network with multiple backup opportunities.

Finally, CECo's meteorological consultants provide a 24 hour per day/7 days per week data source consisting of all routinely available National Weather Service information plus the CECo network data.

The backup system priority is summarized in Tables 3 and 4. For example, if a ground level release occurred (Table 3) at Quad Cities and the primary wind or differential temperature (196 ft. and 196-33 ft.) were lost then the immediate backup measurements would be the second level (296 ft. and 296-33 ft.) at Quad Cities. The backup identified in the table with 'f' represents values provided by the meteorological consultant. Backup systems for accidental elevated releases are shown in Table 4 with similar interpretations.

The backup priority was developed from the following considerations:

(a) As described in Appendix C the sensors and signal conditioners for each elevation on the tower are isolated from one another to the extent possible into two independent paths (denoted by A and B in Figure 1). Since all the towers have wind instruments, at at least two elevations, each is placed in separate paths. Similarly, where multiple measurements of the same parameter are made, they are separated into two paths.

(b) The primary measurements are those located on the tower at the elevation that most appropriately represents the principal release points, i.e. elevated or ground level.

(c) The first backup system for reach release level will come from the signals provided in the alternate path on the same tower.

(d) Data from additional tower systems in the network or from the meteorological consultant comprise the remainder of the backup.

Appendix D describes the findings of research performed for Commonwealth Edison to document atmospheric dispersion in Northern Illinois and to determine the amount of on-site meteorological equipment required at nuclear power plants sited here. This research is pertinent to using the backup network concept.

In 1972, U.S. AEC Safety Guide 23 - Onsite Meteorological Programs (now Regulatory Guide 1.23) was issued to provide guidance to nuclear power plant licensees on suitable onsite meteorological programs. The onsite program was considered necessary in order to provide:

(1) A conservative and realistic assessment of dispersion of radioactive material from, and the radiological consequences of, a spectrum of accidents.

(2) An assessment of the potential annual radiation dose to the public resulting from routine releases.

(3) An assessment of other than radiological environmental effects, such as fogging and icing.

(4) A rapid, conservative assessment of the radiological consequences of an accidental release of radioactive material to the atmosphere.

In 1973, with these requirements of Regulatory Guide 1.23 in mind, CECo proposed to perform research to determine the type of meteorological program required for nuclear stations sited in Northern Illinois. It was believed that the requirement to maintain meteorological programs at all CECo nuclear stations was unnecessary because the topographical and meteorological conditions of this region seemed uniform, with the possible exception of areas adjacent to and affected by Lake Michigan. The study's objective were:

(1) To determine whether or not on-site preoperational meteorological data are needed to formulate an accurate judgement about the dispersive properties of a site; and

(2) To determine if a network of off-site instrumented towers provides adequate data for use during emergencies which might arise during plant operation.

The results of this study, which are included in their entirety in Appendix D, document that conclusions reached regarding a site's dispersive capacity inferred from N site years of preoperational on-site studies are no different from conclusions reached using data from N site years at operating sites in other locations in northern Illinois. Thus, a preoperational program at a new site was judged unnecessay. However, it was found that variations in meteorological conditions are such that off-site data alone are usually insufficient for adequate documentation of an emergency. The evidence does show that meteorological information from a nearby nuclear site can be useful to a site whose tower is inoperable and that an offsite tower can operate as a backup system.

Table 2a

Braidwood Station

Instrument Locations and Data Record

Measurement	Type	Location	Elevation (Above Grade)	Recording Frequency	Recorder Type	Chart Speed	Chart Period
Wind speed/direction	MRI Model 1074-2	Tower	34 ft.	Continuous	Belfort/ Esterline Angus	3"/Hr.	2 weeks
Wind speed/direction	MRI Model 1074-2	Tower	203 ft.	Continuous	"	3"/Hr.	2 weeks
Ambient Air Temper-ature	MRI Model 832	Tower	30 ft.	1 minute	Esterline Angus E1124	3"/Hr.	2 weeks
Differential Temper-ature	MRI Model 832	Tower	199-30 ft.	1 minute	"	3"/Hr.	2 weeks
Dew Point Temperature	EG&G 220	Tower	30', 199'	1 minute	"	3"/Hr.	2 weeks
Precipitation	MRI Model 302 Tipping Bucket	Ground	3 ft.	Continuous	"	3"/Hr.	2 weeks

Table 2b

Byron Station

Instrument Locations and Data Record

Measurement	Type	Location	Elevation (Above Grade)	Recording Frequency	Recorder Type	Chart Speed	Chart Period
Wind speed/direction	MRI Model 1074-2	Tower	30 ft.	Continuous	Belfort/ Esterline Angus	3"/Hr.	2 weeks
Wind speed/direction	MRI Model 1074-2	Tower	250 ft.	Continuous	"	3"/Hr.	2 weeks
Ambient Air Temperature	MRI Model 832	Tower	30 ft.	1 minute	Esterline Angus E1124	3"/Hr.	2 weeks
Differential Temperature	MRI Model 832	Tower	250-30 ft.	1 minute	"	3"/Hr.	2 weeks
Dew Point Temperature	EG&G 220	Tower	30', 250'	1 minute	"	3"/Hr.	2 weeks
Precipitation	MRI Model 302 Tipping Bucket	Ground	3 ft.	Continuous	"	3"/Hr.	2 weeks

Table 2c

Dresden Station

Instrument Locations and Data Recorded

Measurement	Type	Location	Elevation (Above Grade)	Recording Frequency	Recorder Type	Chart Speed	Chart Period
Wind speed/direction	Teledyne/Geo. Tech Series 50	Tower	35 ft.	Continuous	Esterline Angus Series	3"/Hr.	2 weeks
Wind speed/direction	Teledyne/Geo. Tech Series 50	Tower	150 ft.	Continuous	"	3"/Hr.	2 weeks
Wind speed/direction	Teledyne/Geo. Tech Series 50	Tower	300 ft.	Continuous	"	3"/Hr.	2 weeks
Ambient Air Temperature	EG&G Model 110S-M	Tower	35 ft.	1 minute	Esterline Angus E1124	2"/Hr.	3 weeks
Differential Temperature	EG&G Model 110S-M	Tower	150-35 ft.	1 minute	"	2"/Hr.	3 weeks
Differential Temperature	EG&G Model 110S-M	Tower	300-35 ft.	1 minute	"	2"/Hr.	3 weeks
Dew Point Temperature	EG&G Model 110S-M	Tower	35', 150', 300'	1 minute	"	2"/Hr.	3 weeks
Precipitation	MRI Model 302	Shelter Roof	10 ft.	Continuous	Esterline Angus MS401	1.5 cm/Hr.	3 weeks

Table 2d

LaSalle County Station

Instrument Locations and Data Recorded

Measurement	Type	Location	Elevation (Above Grade)	Recording Frequency	Recorder Type	Chart Speed	Chart Period
Wind speed/direction	MRI Model 1022 S&D	Tower	200 ft.	Continuous	Esterline Angus Model 1102S	3"/Hr.	2 weeks
Wind speed/direction	MRI Model 1022 S&D	Tower	375 ft.	Continuous	"	3"/Hr.	2 weeks
Ambient Air Temperature	MRI Model 15021	Tower	33 ft.	1 minute	Esterline Angus Model E1124E (multipoint)	3"/Hr.	2 weeks
Differential Temperature	MRI Model 15021	Tower	200-33 ft.	1 minute	"	3"/Hr.	2 weeks
Differential Temperature	MRI Model 15021	Tower	375-33 ft.	1 minute	"	3"/Hr.	2 weeks
Dew Point Temperature	EG&G 110-SM	Tower	33', 200'	1 minute	"	3"/Hr.	2 weeks
Precipitation	MRI Model 302 Tipping Bucket	Shelter Roof	10 ft.	Continuous	"	3"/Hr.	2 weeks

Table 2e

Quad Cities Station (South)

Instrument Locations and Data Recorded (a)

Measurement	Type	Location	Elevation (Above Grade)	Recording Frequency	Recorder Type	Chart Speed	Chart Period
Wind speed/direction	Climet	Tower	196 ft.	Continuous	Esterline Angus	2"/Hr.	2 weeks
Wind speed/direction	Climet	Tower	296 ft.	Continuous	"	2"/Hr.	2 weeks
Ambient Air Temperature	Rosemont #78-0065-0041	Tower	33 ft.	2 minutes	Esterline Angus E1124	2"/Hr.	2 weeks
Differential Temperature	Rosemont	Tower	196-33 ft.	2 minutes	"	2"/Hr.	2 weeks
Differential Temperature	Rosemont	Tower	296-33 ft.	2 minutes	"	2"/Hr.	2 weeks
Dew Point Temperature	EG&G	Tower	33 ft.	2 minutes	"	2"/Hr.	2 weeks
Precipitation	MRI Model 302 Tipping Bucket	Shelter Roof	10 ft.	Continuous	"	2"/hr.	2 weeks

(a) In addition there is an MRI Series 10-22 wind speed/direction sensor on a 30 ft. pole located in the switchyard for providing wind information to the control room on an interim basis. This system will be discontinued when the ODCS is fully operational.

Table 2f

Zion Station

Instrument Locations and Data Recorded

Measurement	Sensor Type	Location	Elevation (Above Grade)	Recording Frequency	Recorder Type	Chart Speed	Chart Period
Wind speed/direction	Teledyne 1500 Series	Tower	35 ft.	Continuous	Esterline Angus L1102S	3"/Hr.	2 weeks
Wind speed/direction	"	Tower	125 ft.	Continuous	"	3"/Hr.	2 weeks
Wind speed/direction	"	Tower	250 ft.	Continuous	"	3"/Hr.	2 weeks
Ambient Air Temperature	Bristol Shielded Resistance Thermometer	Tower	250-35 ft.	1 minute	Westronics multipoint Model M11D2	2"/Hr.	3 weeks
Differential Temperature	Bristol Shielded Resistance Thermometer	Tower	125-35 ft.	1 minute	"	2"/Hr.	3 weeks
Differential Temperature	Bristol Shielded Resistance Thermometer	Tower	250-35 ft.	1 minute	"	2"/Hr.	3 weeks
Dew Point Temperature	Foxboro Dewcell	Instrument Shelter	5 ft.	1 minute	"	2"/Hr.	3 weeks
Precipitation	MRI Model 302 Tipping Bucket	Shelter	10 ft.	Continuous	"	2"/Hr.	3 weeks

Table 3

Backup Metro Measurements Program
Ground Level Release(a)

Station	Primary*		Backup		Tertiary		4th		5th	
	Wind	ΔT	W	ΔT	W	ΔT	W	ΔT	W	ΔT
Braidwood (Bd)	1	2	2	f	D1	D2	L2	L2	f	f
Byron (By)	1	2	2	f	f	f	Rockford**		-	-
Dresden (D)	1	2	2	3	Bd1	Bd2	L2	L2	3	f
Quad Cities (Q)	2	2	3	3	f	f	Moline**		-	-
Zion (Z)	1	2	2	3	3	f	f	f	-	-

a the levels are numbered from the lower level up the tower; level 1 is typically at a height of 35'. Levels 1, 2, 3 only are available to the plant's process computer and thus, to the Control Room. All listed measurement systems are available to the SYFA and IBM 3033 computers and thus to the plant's TSC and corporate office.

* information for any group must come from same source; i.e., one can't mix stations; ex. D1 Bd2. ΔT represents stability class.

f hindcast, nowcast and forecast for station

** National Weather Bureau stations - information that could be provided to a station by the meteorological contractor.

Table 4

Backup Metro Measurements Program
Elevated Release
- Level -a

Station	Primary* Wind	Primary* ΔT	Backup W	Backup ΔT	Tertiary W	Tertiary ΔT	4th W	4th ΔT	5th W	5th ΔT
Dresden	3	3	2	2	Bd2	Bd2	L3	L3	L2	L2
LaSalle	3	3	2	2	D3	D3	Bd2	Bd2	f	f
Quad Cities	3	3	2	2	f	f	Moline**			

a the levels are numbered from the lower level up the tower; level 1 is typically at a height of 35'. Levels 1, 2, 3, only are available to the plant's process computer and thus, to the Control Room. All listed measurement systems are available to the SYFA and IBM 3033 computers and thus to the plant's TSC and corporate office.

* information for any group must come from same source; i.e., one can't mix stations; ex. D1 Bd2. ΔT represents stability class.

f hindcast, nowcast and forecast for station

** National Weather Bureau stations - information that could be provided to a station by the meteorological contractor.

3. Weather Forecasts

Forecasts will be prepared by CECo's meteorological consultant* routinely twice each day. Each forecast is for a 36-hour period beginning at 1200 CST or 2400 CST.

The hourly forecasted parameters include the following:

All Sites:	Contractor's Forecasted Input	Output to CECo
1.1 1-Wind Speed	KTS, MPH	MPS
1.2 1-Wind Direction	Degrees	Degrees
1.3 1-Stability	Stab.Class	DeltaT/DeltaZ

ZION Only:

1.4 1-Air Temp. over water	°F	0 - for no lake effect**
1.5 1-Air Temp. over land	°F	
1.6 1-Air Mass Stability	DeltaT/DeltaZ	1 - for Case 1 lake effect
		2 - for Case 2 lake effect

The corporate SYFA will poll the consultant's computer every 12 hours automatically. The plant SYFA will poll the corporate IMB computer on command, at least twice a day at scheduled intervals. In addition the plant SYFA could call the consultant's computer for the forecast should communications with the corporate office be interrupted.

The corporate ODCS operator and the TSC ODCS operator may use these weather forecasts to estimate radiation doses accruing from postulated future releases of radioactivity.

4. Lake Effects at Zion Station

Currently recommended meteorological programs and diffusion methods for nuclear power plants located in coastal zones were recently reviewed for the U.S. Nuclear Regulatory Commission (NUREG/CR-0936 BNL-NUREG-51045, October 1979). Among certain deficiencies in guidelines and procedures noted in this document were "failure to consider the role of coastal internal boundary layers, specifications for tower locations and instrument heights, (and) methods for classifying atmospheric stability...". Included were recommendations for changes to the guidelines.

An atmospheric dispersion model has been developed to account for boundary layer conditions that could occur at the Zion plant. The model development essentially followed the various methods itemized

*Currently Murray and Trettel, Inc., Northfield, Illinois.
**See next section for description of lake effect conditions.

in the reference cited. Conservatively high ground level concentrations result from the model when compared to standard dispersion calculations. (Section 9.4 of Appendix B provides additional information on the lake breeze model.)

The Boundary Layer

Continuous measurements of the boundary layer in the vicinity of Zion are not available. Indeed, aside from a few intensive short term studies of lake shore dispersion in the vicinity, no boundary layer data exist. Consequently readily available meteorological measurements representing a two year period were used in conjunction with the boundary equation (1) found in NUREG-0936 to infer the existence and location of the boundary.

NUREG-0936 equation (1) was evaluated subject to the following assumptions and conditions:

(1) friction velocity $U^* = 1$ mps
(2) Wind speed less than or = 6 mps
(3) Land-water temperature contrast at least 5^oF
(4) Air mass stability was estimated by the 250-125 ft. differential temperature measured on the Zion tower.
(5) Wind direction onshore

The results are shown in Figure 2. In summary, the boundary was computed to occur roughly 10 percent of the hours annually (876/8760). Of those 876 hours it occurred well above the Zion ventilation stacks 95 percent of the time (832 hours). The remaining 5 percent of the time (44 hours) it was below the stacks leading to potential fumigation downwind (cf. Figure 3).

It should be noted that the existing Zion meteorological tower is located entirely within the calculated boundary. For all practical cases, then, the measurements from the tower can be assumed to represent the boundary layer conditions and not be partway in the boundary layer and partway in the 'lake' air (a caution referred to in NUREG-0936).

Dispersion Model

When a boundary height, variable both in time and inland fetch, is taken into account, four downwind zones with different dispersion characteristics emerge. The dispersion equations differ for the four cases summarized below.

Case 1. The boundary layer is located above the stacks. Consequently vertical dispersion is limited by the boundary and the ground at all ranges downwind to 10 miles (the downwind extent of the model evaluation). Boundary layer dispersion is characterized by meteorological tower measurements.

Case 2. The boundary layer is located below the stacks. This can lead to three distinct cases depending on the downwind range in question.

Case 2.1. At downwind distances from the stacks to the point X_1, beneath which the bottom of the plume intersects the boundary. The plume is embedded in the relatively turbulent-free lake air.

Case 2.2. At downwind distances from point X_1, to the point X_2, beneath which the top of the plume intersects the boundary. In this zone fumigation is assumed to occur. The effluent is uniformly distributed in the vertical.

Case 2.3. At downwind distances beyond the point X_2. Here limited mixing occurs due to the plume being trapped beneath the boundary. Here also the effluent is uniformly distributed in vertical.

Results

The model was evaluated at various downwind distances to ten (10) miles, to yield the 'worst case' values. The highest concentrations were due either to Case 1 or Case 2.2. The remaining cases were therefore eliminated as possible worst cases.

Required Forecast Inputs to Model

The lake effects model requires a variety of inputs. Some are used to determine whether or not a boundary exists. Others are used to select the limited mixing or the fumigation mode. The inputs used to decide whether lake effects will occur are:

Hour of day
Wind Direction
Wind Speed
Temperature contrast between lake and land

The additional input used to select the appropriate dispersion mode is air mass stability.

Signals representing the temperature differential between lake and land and air mass stability are not directly available. Instead they are determined from a variety of meteorological reporting stations and provided by the meteorological consultant. Predicted hourly differential and stability factors are also prepared by the consultant.

The presence or absence of a lake effect condition will be reported by the meteorological consultant and appended to the Zion Station forecast. A "zero" (0) will indicate that no lake effect condition is forecasted for a particular hour. A "one" (1) will indicate that there is a forecasted Case 1 lake effect condition. A "two" (2) will indicate that there is a forecasted Case 2 condition. Using this lake effect indicator the computers will chose the appropriate atmospheric dispersion model for estimating the offsite consequences of a release. Appendices A and B describe this further.

Figure 2

ZION STATION

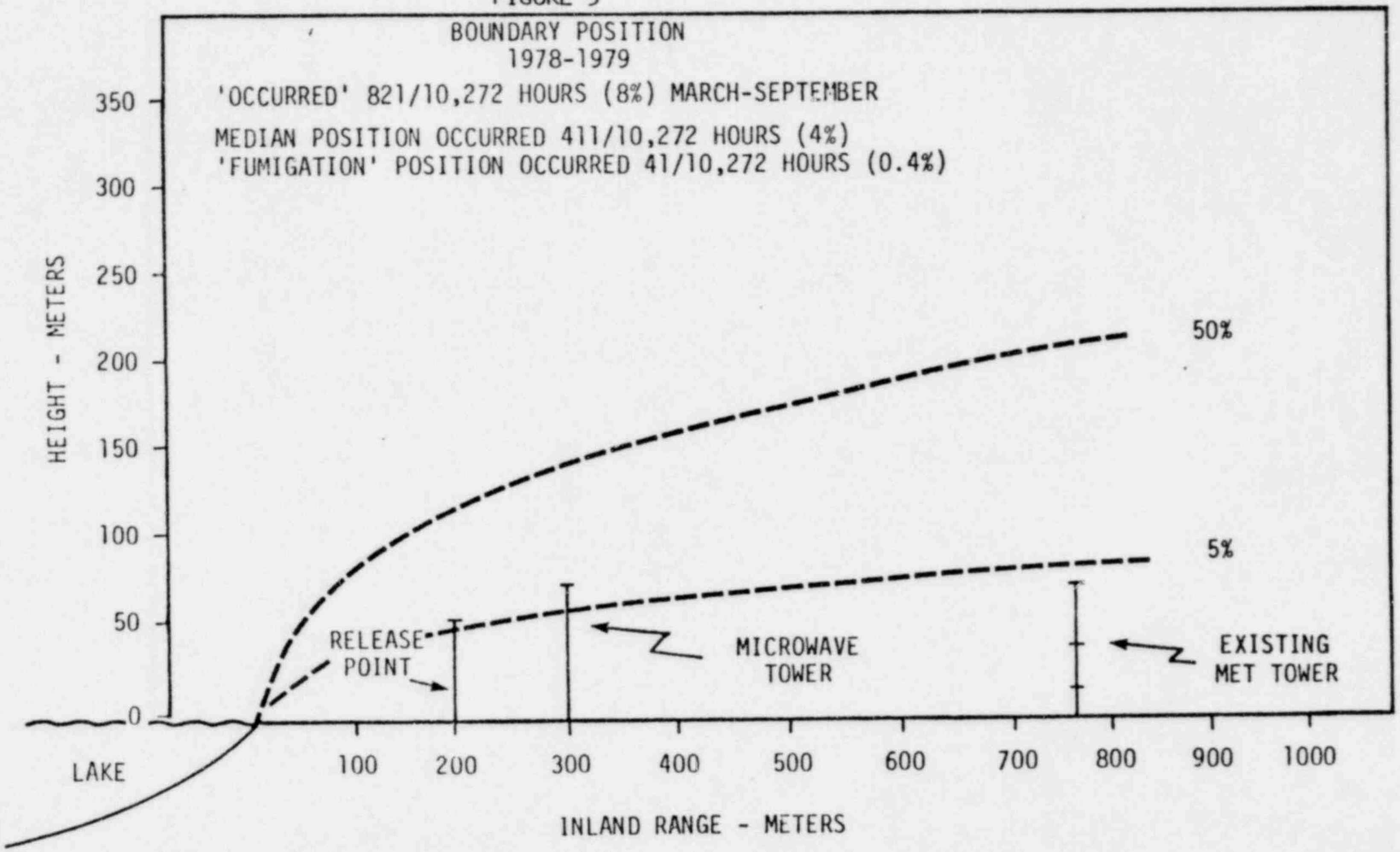
Estimated Frequencies of Occurence*

(Hours per year - Percent)

. No Lake Effects	90
. Lake Effects	10
	100
. Lake Effect Trapping	9
. Lake Effect - Fumigation	1
	10

* Based on 1978 - 1979 Hourly Measurements
(March through November)

FIGURE 3


BOUNDARY POSITION
1978-1979
'OCCURRED' 821/10,272 HOURS (8%) MARCH-SEPTEMBER
MEDIAN POSITION OCCURRED 411/10,272 HOURS (4%)
'FUMIGATION' POSITION OCCURRED 41/10,272 HOURS (0.4%)
350
300
250
200
150
100
50
0
HEIGHT - METERS
50%
5%
RELEASE
POINT
MICROWAVE
TOWER
EXISTING
MET TOWER
LAKE
100
200
300
400
500
600
700
800
900
1000
INLAND RANGE - METERS

5. Nuclear Regulatory Commission Nuclear Data Link (NDL)

The NRC staff is engaged in improving the capabilities of its NRC Operations Center (OC) at Bethesda, Maryland. One aspect of this effort involves the transmission of various plant parameters including meteorological data over the NDL from each nuclear plant to the OC. When the scope of the NDL has been more fully developed by the NRC CECo will review the NRC specifications for application to Zion Station and elsewhere.

6. Atmospheric Release Advisory Capability (ARAC)

In April 1980 FEMA asked CECo to participate in a two year pilot program to test the application of ARAC to nuclear stations. Zion Station was to be a test facility, along with the Indian Point Station in New York. (ARAC is a Lawrence Livermore Laboratory computer system for estimating the regional consequences of an accidental release of radioactivity.) If CECo ever participates in this program, the ARAC mini-computer, printer, terminal and TV-screen will be placed not at Zion but at the corporate office and operated off the meteorological feed from the SYFA computer. At this time, however, due to severe financial difficulties CECo is not in a position to participate in the ARAC pilot study. If this situation changes this decision will be re-evaluated.

IV. Model Accuracy and Conservatism

Commonwealth Edison has adopted for use the atmospheric transport and plume gamma dose models recommended by the Nuclear Regulatory Commission in its Regulatory Guide series (e.g., RG 1.23, 1.109 and 1.111) and in the publication "Meteorology and Atomic Energy 1968" (TID-24190, July 1968). Discussions of the accuracy and conservation of these models are scattered throughout the published literature, a sampling of which is given in Appendix E.

Two very relevant documents to Commonwealth Edison are references 8 and 9 in Appendix E. Reference 8 is a state-of-the-art review of meteorological measurement and atmospheric transport and diffusion prediction models for plants located in coastal zones, such as Zion Station. Whereas this study by Brookhaven National Laboratory was restricted to simple coastlines (such as near Zion) without complex terrain, that only effects within five miles of the plant should be considered, and that models recommended should give conservative predictions for plant design purposes, CECo has adopted the model to the realtime prediction situation as was discussed in Section III.4 and Appendix B of this report.

This modified model should be adequate for the purpose intended: to help the control room operator and the ODCS operator reach a decision concerning the necessity to recommend protective actions in the vicinity of the plant during the initial phase of an accident, i.e., before field personnel are fully capable of tracking the direction of and measuring the radiation intensity from the plume, and to make a reasonably conservative estimate of radiation dose to the public. Once field personnel are dispatched and the plume's behavior is being tracked from the ground and/or air, then the role of a predictive meteorological model is reduced.

Appendix E reference 9 reviews the uncertainty in atmospheric dispersion models to 50 miles. Tables 3 to 6 reproduced herein from reference 9 summarize the uncertainty associated with concentration predictions made by the Gaussian plume atmospheric dispersion model. CECo does not disagree with these findings, in fact our own research supports the accuracy estimates for locations near the plant.

Appendix F contains the results of recent research sponsored by CECo to determine the validity for elevated releases of the finite plume gamma dose and Gaussian diffusion models, respectively. The findings on the finite plume model were: (1) that the most commonly used finite plume model, the sector average model, underestimates the actual plume exposure by aproximately 60%, (2) that the more complicated, off-axis model from Eq. 7.43 of "Meteorology and Atomic Energy 1968" was more accurate than the sector average model and overestimated the measured exposure by a factor of 20% to 40% (3) that the accuracy of the off-axis plume model could be enhanced if atmospheric stability classes were constrained to the neutral and stable categories, and (4) that wind direction measured at an intermediate height (rather than at 100 m) may better represent the center line of an elevated plume during extremely unstable atmosperhic conditions.

IMAGE EVALUATION TEST TARGET (MT-3)

6″

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

In another research program sponsored by CECo the ability to predict SO_2 concentrations was tested. A sulfur dioxide emissions control program was operated during 1976 - 1979 at CECo's Powerton Plant, an 1800 Mw (electric) generating plant located on the Illinois River south of Peoria, Illinois. Gaseous emissions were limited when SO_2 concentrations could exceed applicable standards. A Gaussian plume model was used to compute ground level concentrations from meteorological and plant inputs. A 13-site SO_2 monitoring network provided measured hourly concentrations in the vicinity. These data permitted direct comparisons with the model output. A detailed model evaluation was therefore possible.

The atmospheric dispersion model was adapted from the Bierly - Hewson limited mixing equation. Total reflection at both the ground and at the top of the mixed layer was accounted for. The height of the mixed layer was measured continuously with monostatic and bistatic acoustic sounders. Horizontal and vertical dispersion coefficients were based on the Pasquill-Gifford curves given in Turner's Workbook. Atmospheric stability was inferred from surface observations taken hourly at the nearby Peoria Municipal Airport (PIA).

A one year record (July 1, 1976 through June 30, 1977) was used to compare the modeled SO_2 results with the measured concentrations. Actual plant emission rates were used in the model. Meteorological inputs were actual measurements from PIA or from onsite instrumentation. Figures 4, 5 and 6 illustrate the highest 5% of the predicted values versus the observed SO_2 values at an urban site located approximately 12 km north of the plant. Asterisks represent the 1:1 ratio of predicted and observed values for 1-hour (Figure 4), 3-hour (Figure 5) and 24-hour (Figure 6) running averages. The modeled (predicted) values were higher than the measured (observed) values on all three averaging periods, but the linear correlation increased dramatically with the increased averaging periods.

Table 3 An estimate of the uncertainty associated with concentration predictions made by the Gaussian plume atmospheric dispersion model[a]

Conditions	Range of the ratio $\frac{\text{Predicted}}{\text{Observed}}$
Highly instrumented flat-field site; ground-level centerline concentration within 10 km of continuous point source	0.8-1.2
Specific hour and receptor point; flat terrain, steady meteorological conditions; within 10 km of release point	0.1-10
Ensemble average for a specific point, flat terrain, within 10 km of release point (such as monthly, seasonal, or annual average)	0.5-2
Monthly and seasonal averages, flat terrain 10-100 km downwind	0.25-4
Complex terrain or meteorology (e.g., sea breeze regimes)	[b]

[a]T. V. Crawford (Chairperson), Atmospheric Transport of Radionuclides, pp. 5-32 in Proceedings of a Workshop on the evaluation of Models Used for the Environmental Assessment of Radionuclide Releases, ed. by F.O. Hoffman, D. L. Shaeffer, C. W. Miller, and C. T. Garten, Jr., USDOE Report CONF-770901, NTIS, April 1978.

[b]The group which assembled these estimates did not feel there was enough information available to make even a "scientific judgment" estimate under these conditions.

Table 4 Some validation results for ensemble averages predicted by the Gaussian plume model

Conditions	Range of the ratio $\frac{\text{Predicted}}{\text{Observed}}$
Annual average SO_2 concentrations for Roane Co., Tennessee; both point and area source emissions included	0.5-$\leq$2
Continuous gamma-ray measurements 0.04-6.8 km downwind of a boiling water reactor	0.33-1.78
Gamma-ray doses downwind of Humboldt Bay Nuclear Power Plant	0.5-$\leq$2
Monthly gamma-ray doses for four stations downwind of a nuclear power plant at an inland site	0.30-4.78 individual stations 1.55 mean of all data

Table 5 Validation results for Gaussian plume model predictions out to 140 km

Conditions	Range of the ratio Predicted/Observed
^{85}Kr measurements 30-140 km downwind of the Savannah River Plant	
Weekly and annual averages	0.25-4
Seasonal averages, Spring	2-4, 69% of samples 2-10, 100% of samples
Summer	0.5-4, 46% of samples 0.5-10,85% of samples
Fall	0.5-4, 31% of samples 0.5-10, 85% of samples
Winter	2-4, 69% of samples 2-10, 92% of samples
Annual Average	1-4, 77% of samples 1-10, 92% of samples
10-hour averages, six variations of the model	0.5-2, 42-65% of samples 0.1-10, 79-95% of samples

Table 6 Some validation results for Gaussian plume model predictions in speed, inversion conditions both complex terrain and also under low wind

Conditions	Range of the ratio Predicted/Observed		
Review of a number of experiments conducted in complex terrain for plume centerline concentrations	0.01-300, individual measurements close to the source		
	0.50-2, <2-15 km downwind of source		
Review of a number of experiments conducted under low wind speed, inversion conditions			
	stability category		
	E	F	G
smooth desertlike terrain[a]	2.3-10	1.3-12	3.6-20
wooded flat terrain[a]	20-25	20-40	20-30
wooded hilly terrain[a]	50-350		300-500

[a]Ratios estimated from curves provided by Van der Hoven.[41]

FIGURE 4

PLOT OF HIGHEST 5% OF 1 HR.SO2 CONC. FOR THE STARR SITE

PLOT OF OBSERVED VS PREDICT

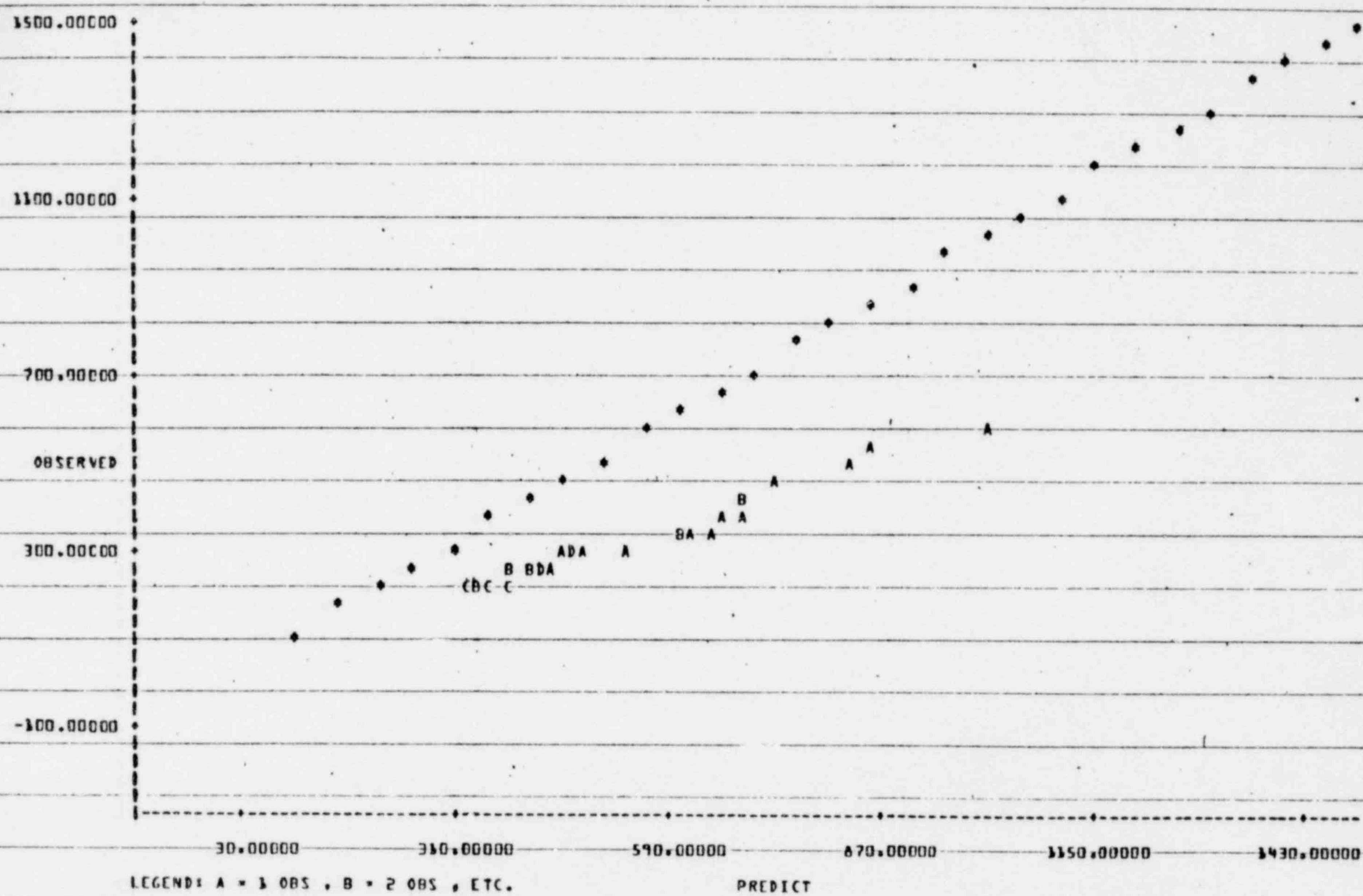


FIGURE 5

PLOT OF HIGHEST 5% OF 3 HR.SO2 CONC. FOR THE STARR SITE

PLOT OF OBSERVED VS PREDICT

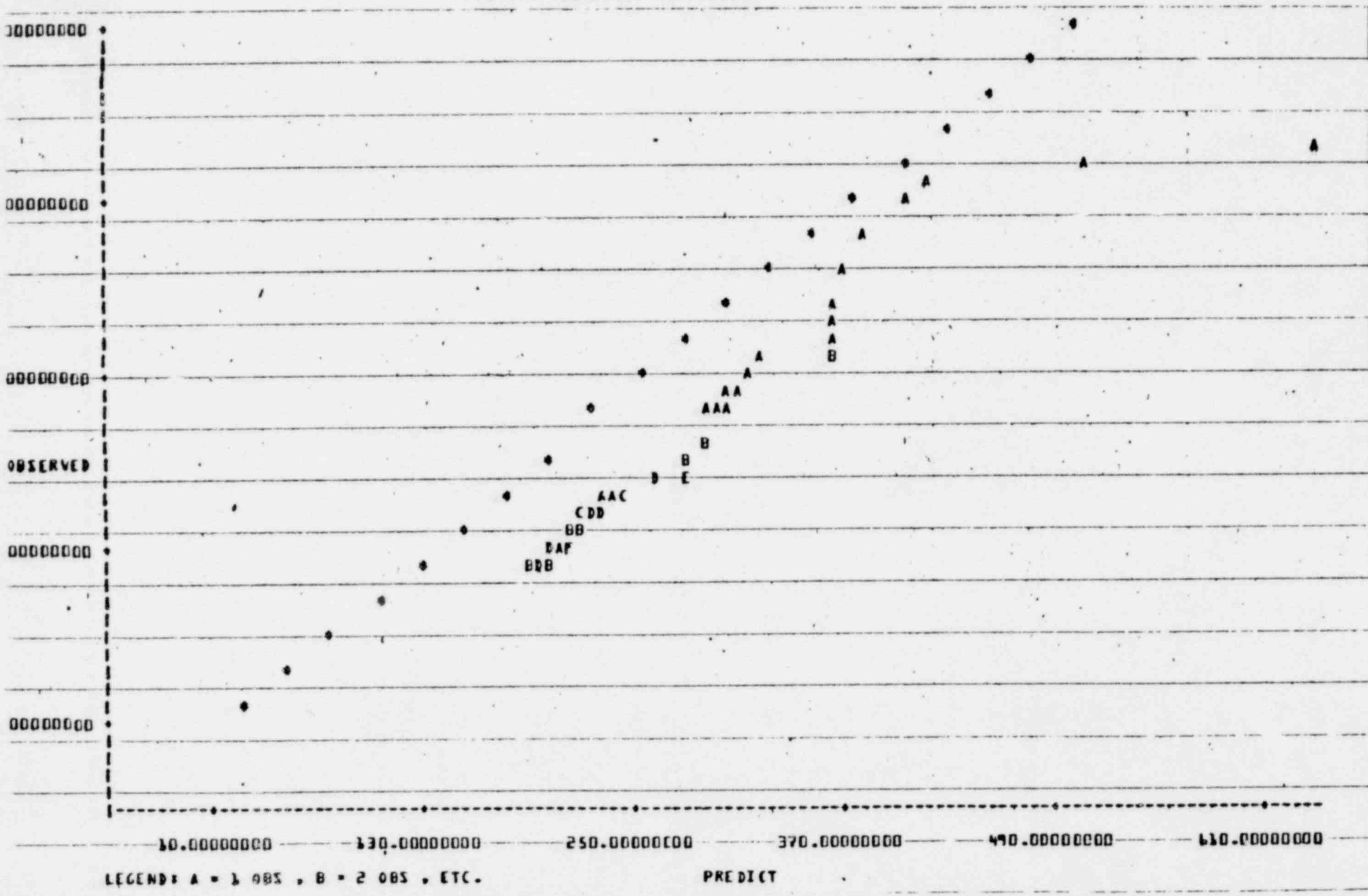
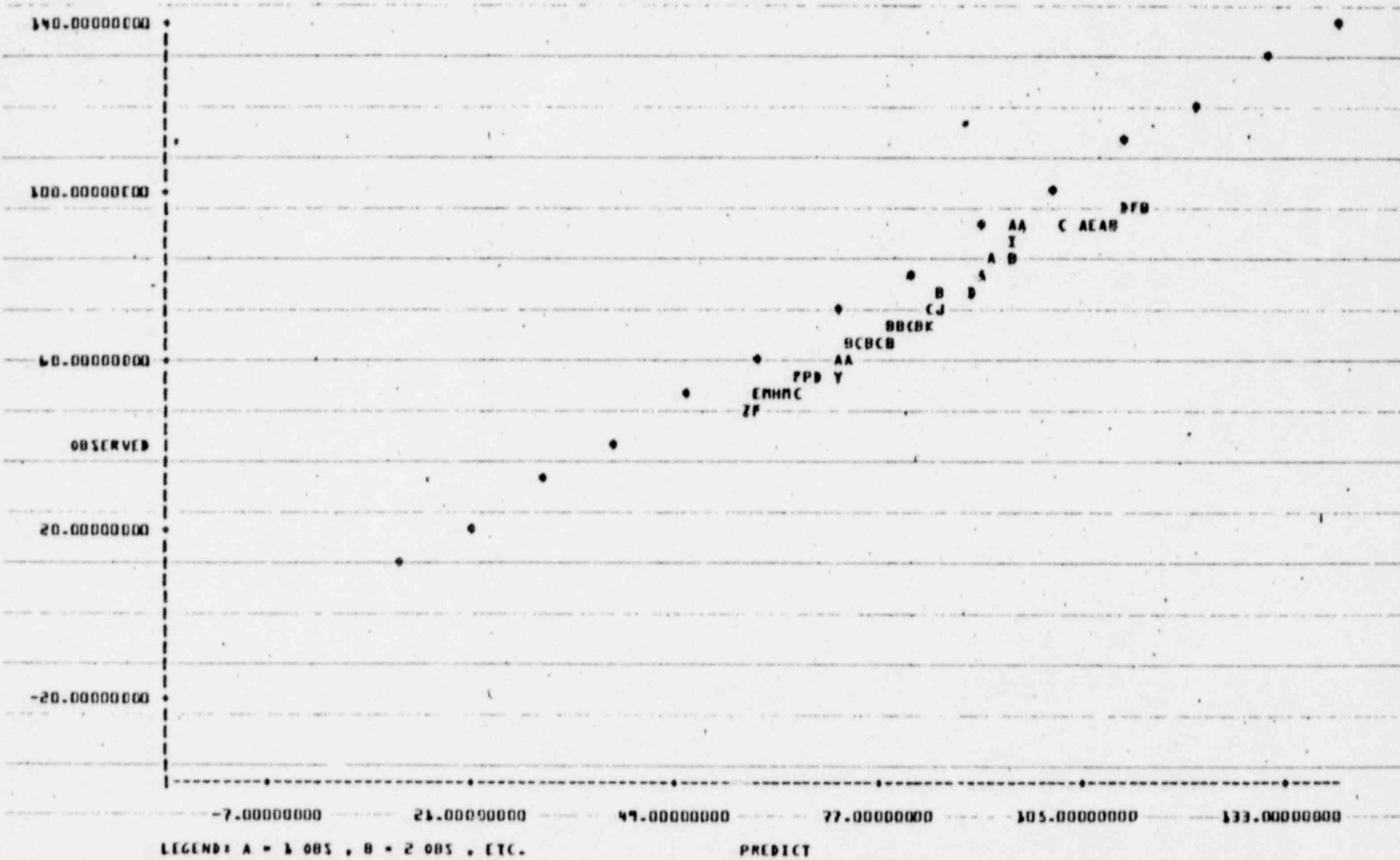


FIGURE 6

PLOT OF HIGHEST 5% OF 24 HR. SO2 CONC. FOR THE STARR SITE

PLOT OF OBSERVED VS PREDICT



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V. Quality Assurance Program

The NRC Zion Order dated 2/29/80 and NUREG-0654 Appendix 2 requires the establi ment of a quality assurance program (Q.A.P.) consistent with applicable provisions of Appendix B to 10 CFR 50. It states further that the acceptance criteria stated in Revision 1, Section 17.2 of NUREG-75/087 apply. CECo agrees that a Q.A.P. can be developed consistent with applicable provisions of 10 CFR 50, Appendix B. The Commonwealth Edison Company disagrees with the NRC on its position that all of the provisions of Section 17.2 can be applied to the meteorological program, a professional consulting service.

The Commonwealth Edison Company has had a formal quality assurance program for its meteorological monitoring since 1976. The scope of the Q.A.P. is delineated in Standard Quality Assurance Articles which are appended to the contract specifications. The current Articles (Rev. 0) and current Q.A.P. are provided in Appendix G of this report. The Q.A. Articles for meteorological monitoring were adopted specifically for this program from 10 CFR 50 Appendix B. However, since the meteorological facility is not composed of structures, systems and components that prevent or mitigate the consequences of postulated accidents and is thus not "safety related", not all aspects of 10 CFR 50 Appendix B apply. Those aspects of quality assurance germane to supplying good meteorological information for a nuclear power plant were kept in the Articles and incorporated into the contractor's Q.A.P.

Commonwealth Edison has reviewed the requirements for the proposed off-site dose calculation system with respect to the applicability of 10 CFR 50 Appendix B and has identified certain areas wherein strengthening of the existing Q.A. Articles is appropriate. It is our intention to incorporate such extended Q.A. Articles, a copy of which is contained in Appendix G also, in the forthcoming program. Since other quality needs may emerge from our current reviews, Commonwealth Edison will continue its investigations, but in any event will incorporate such Articles in its program by 1/1/81.

VI. Schedule

The ODCS implementation schedule is:

(1) Provide a weather forecast to Zion Station by August 29, 1980 (6 months after the NRC order was issued).

(2) By 8/29/80 be able to poll by computer all meteorological towers, including the two non-operating stations, Byron and Braidwood.

(3) Install the basic ODCS and forecast at Zion, Dresden, Quad Cities, and LaSalle County Stations as soon as possible but no later than 1/1/81. The basic ODCS is shown heavily lined in Figure 7.

(4) Install the full ODCS including redundant power supply to each meteorological tower at the four operating nuclear stations by 12/31/81. The IDPH computer network shown in the upper right corner of Figures 1 and 7 is not a component of the ODCS but is included to show how meteorological information is provided to the State.

(5) Implement an upgraded QA program by 1/1/81.

(6) Install the Nuclear Data Link as soon as practical after the NRC specifications are finished.

(7) Before the operating license install full ODCS at Byron and Braidwood.

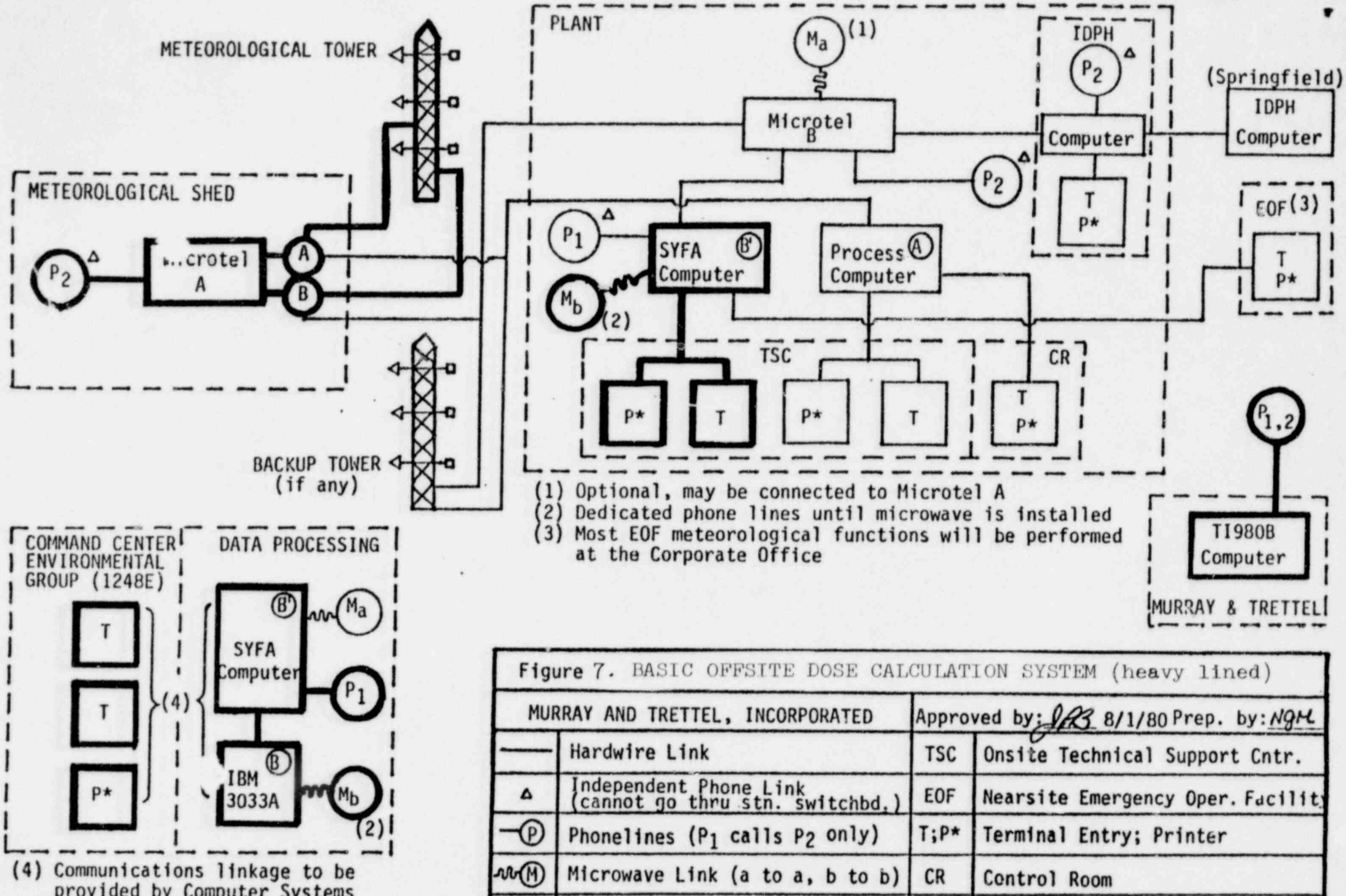


Figure 7. BASIC OFFSITE DOSE CALCULATION SYSTEM (heavy lined)

MURRAY AND TRETTEL, INCORPORATED		Approved by: JRB 8/1/80 Prep. by: NGM	
——	Hardwire Link	TSC	Onsite Technical Support Cntr.
Δ	Independent Phone Link (cannot go thru stn. switchbd.)	EOF	Nearsite Emergency Oper. Facility
—(P)	Phonelines (P_1 calls P_2 only)	T;P*	Terminal Entry; Printer
∿(M)	Microwave Link (a to a, b to b)	CR	Control Room
A,B	Redundant MET tower signal paths	IDPH	Illinois Dept. of Public Health
(A)(B)(B')	Computer Models		

Appendix A

ODCS Class A Model

June, 1980
Rev. 2

Appendix A

ODCS Class A Model

W and Honeywell Process Computer

A. Continuous Polling

(1) Poll meteorological equipment at the following frequencies:

WS/WD every 5 seconds
ΔT and other signals once every minute

(2) Test the meteorological data for invariant signal transmission and issue a warning when the criteria are exceeded.

(3) Calculate 15-minute averages of each parameter four times per standard hour and store 3 hours worth of these averages* in a retrievable file in either CST or DST clock time, as appropriate.**

(4) Provide all meteorological averages, for any level, on command.

B. Continuous Analysis

(1) Compute 2-minute and/or 30-minute averages of noble gas release (uCi/sec) via principal effluent pathways (ground level or elevated) and issue a warning if the following criteria$^{\Delta}$ are met or exceeded.

	$\overline{Q}_2$ (uCi/sec)	$\overline{Q}_{30}$
ground level	8.9×10^6	8.9×10^5
elevated	1.3×10^8	1.3×10^7

(2) If warning is issued provide the last computed $\overline{WS}_{15}$, $\overline{WD}_{15}$, and stability class (a station-dependent function of ΔT_{15}).

(3) The warning messages are:

(a) for $\overline{Q}_2$, the Site Emergency EAL of 500 mR/hr offsite using worst case meteorology has been exceeded.

(b) For $\overline{Q}_{30}$, the Site Emergency EAL of 50 mR/hr offsite using worst case meteorology has been exceeded.

*Be aware that special rules apply for wind direction. These will be provided.

Δ Criteria may change on a station-by-station basis; command to permit manual entry would be useful.

**The computer should display information in CST or DST as appropriate.

C. On Command Analysis (from CR, TSC, or EOC)

(1) Using $\overline{Q}_2$, $\overline{WS}_{15}$, $\overline{WD}_{15}$ $\overline{\Delta T}_{15}$, or other parameter if required, compute continuously (i.e., "running 2-minute averages"):

$$D' (\overline{WD}, R)_{Max} = \frac{\overline{Q}_2 \cdot K (\overline{\Delta T}_{15})_{MAX}}{\overline{WS}_{15}} \text{ mR/hr}$$

where D' is the maximum offsite dose rate for a release of $\overline{Q}_2$, and $K (\overline{\Delta T}_{15})_{max}$ are computer-stored dose factors (to be provided). If $D' \geq 1000$ mR/hr issue a warning.

(2) The warning message is:
The General Emergency EAL of 1000 mR/hr Offsite using actual meteorology has been exceeded.

(3) Accompany each estimate of D' with $\overline{WD}$, $\overline{WS}$, and stability class (a, f (Δ T)), and name the downwind direction.

(4) Compute D' ($\overline{WD}$, R) at each of the 6 pre-selected downwind ranges, one value of which will always include D' $(\overline{WD}, R)_{max}$. Name the affected sector and stability class.

Class A Model

Purpose: To provide initial transport and diffusion estimates within 15 minutes following classification of an incident.

Background: The control room operator will rel-- on the meteorological and effluent release rate information provided by the plant process computer and converted into Emergency Action Levels (EAL) by the Class A computer model. The model will activate the necessary EAL alarms for site emergency and for general emergency.

The Class A Model will activate the necessary EAL alarms for site emergency: (1) 2-minute average noble gas release rate having projected offsite dose rate of 500 mR/hr and (2) 30-minute average noble gas release rate having projected offsite rate of 50 mR/hr., using worst case meteorology and for general emergency: 2-minute average noble gas release rate having projected offsite dose rate of 1000 mR/hr using 15-minute average actual meteorology. These computations will be made using the effluent release and whole body dose factors for the isotope GR-999.

Meteorological data will be hardwired into the system. Wind signals will be sampled every 5 seconds and differential temperature data once every minute. 15 minute averages will be computed four times per standard hour. Each 15-minute average will be tested for invariant signal transmission and a warning issued when those criteria are exceeded.*

The meteorology and effluent release are combined to determine the resulting whole body dose rate at offsite locations.

Procedures: 1. Meteorological Data - 15-minute averages

Sample WS/WD signals every 5 sec.; ΔT signals one every minute; Compute a standard deviation of the wind direction field for the 15 minute period. Compute all 15-minute averages for all parameters.

2. Effluent Data

Compute 2-minute and 30-minute averages of noble gas release (uCi/sec) via principal effluent pathways and issue a warning if criteria are met or exceeded (See Appendix A, Section B). Monitors will provide data in μCi/sec.

*All 15-minute averages will be stored in the process computer for 3 hours.

3) On Command Analysis

Compute max dose rate using $\overline{Q}_2$ and the 15-minute meteorology. If the calculated dose rate meets or exceeds the criteria (Appendix A), a warning will be issued. In addition, the dose rate will be calculated at each of the preselected downwind ranges (one value will be $\underline{D}$ max). Each dose rate will be printed out along with $\overline{WD}$, $\overline{WS}$, stability class, affected (downwind) sector and adjacent downwind sectors (Figure 1).

A. For non-lake breeze conditions (at Zion, L.B. marker=0).

$$D'(\overline{WD},R) = \frac{\overline{Q}_2\, K \exp(-\lambda t)}{\overline{WS}_{15}} \quad \text{mrem/hr (1)}$$

K - factors are found in table 9.2-1 (attached herein for ground level and elevated release.

$\lambda =$ 3.4×10^{-4} Sec.$^{-1}$

t = Range (m) $\div$ $\overline{WS}_{15}$ (m/sec)

B. For Zion Lake breeze cases: Use equation (2)

$$D'(\overline{WD},R) = \overline{Q}_2 \cdot \chi/Q \cdot X \cdot \frac{1}{8760} \cdot C \exp(-\lambda t) \text{ mrem/hr}$$

where χ/Q is relative concentration factor in Table 9.4-2.

X_i is 5.43 E + 03 mrem/yr per uCi/m^3

1/8760 converts 1/yr to 1/hr.

C is the "split sigma" correction factor of Table 9.2-6.

Attach.: Copy of Appendix A
Flow Charts
Notes

CLASS A
MODEL

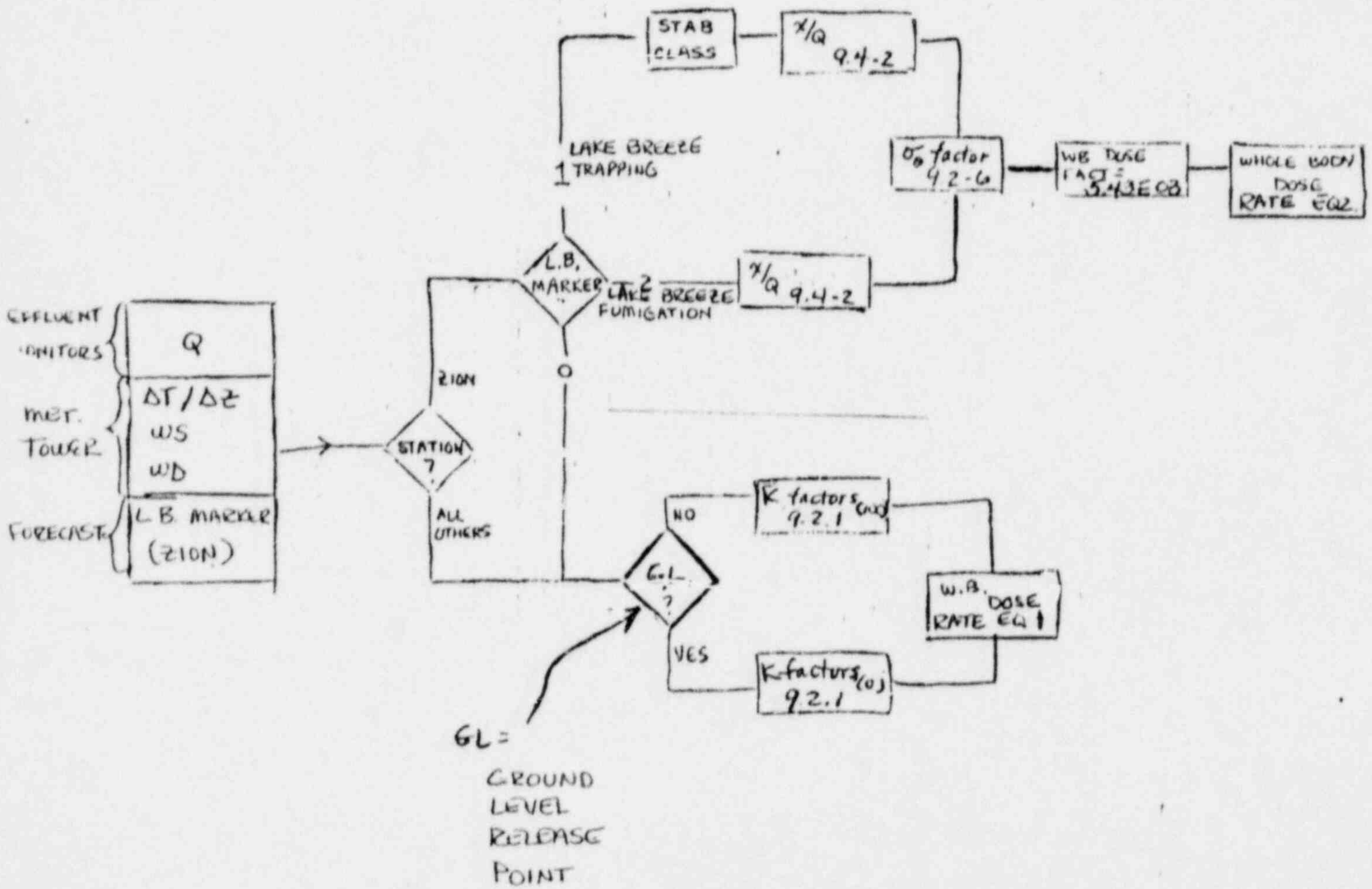


FIGURE 1
ON COMMAND ANALYSIS OF OFFSITE DOSE RATE (mrem/hr)
USING REAL-TIME METEOROLOGY

A. Continuous Polling - (Notes for Appendix A)

1) WS/WD every 5 sec
   ΔT once every minute

The following levels are to be polled for each site:

DNPS 35′ WS, WD, 150ΔT; 300′ WS, WD, ΔT (first set vent, second chimneys
ZION 35′ WS, WD, 125ΔT; 250′ WS, WD, ΔT (first set primary, second backup)
LSCS 375′ WS, WD, ΔT; 200′ WS, WD, ΔT (first set primary, second backup)
BRWD 30′ WS, WD, 199ΔT; 199′ WS, WD, ΔT (see Zion comment)
BYRN 30′ WS, WD, 250ΔT; 250′ WS, WD, ΔT (see Zion comment)
SOQD 196′ WS, WD, ΔT; 296′ WS, WD, ΔT (see Dresden comment)

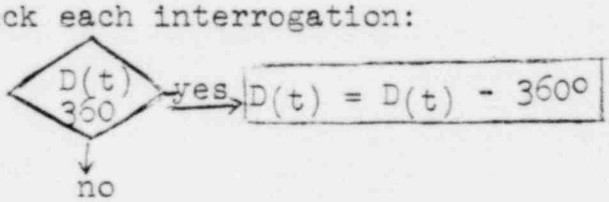
2) 15-minute Average

Calculate 15-minute averages of each parameter four times per standard hour. The minimal acceptable number of ΔT interrogations per fifteen minute period is 8. (They do not have to be consecutive interrogations).

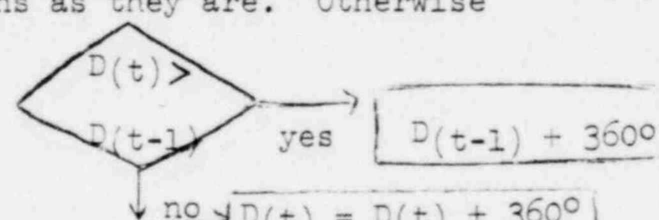
For WS/WD at least 48 interrogations will make up the 15-minute value.

0/540° cross-over:

a) Check each interrogation:



Check the differences between interrogations. If $|D(t) - D(t-1)| \leq 180°$ then accept directions as they are. Otherwise



This is repeated for all directions in each 15-minute average. Store 3-hours worth of 15-minute averages for all parameters in a retrievable file in CST clock time. Recall and display in CST or CDT as appropriate for the time of year.

To compute σ_θ from the WD interrogations use the formula

$$\sigma_\theta = \sqrt{\frac{\Sigma WD^2}{N-1} - \frac{(\Sigma WD)^2}{N(N-1)}}$$

where WD = the value of the interrogation; N = number of interrogations σ_θ should be computed on a 5-minute running average. N must be at least 30.

3) Testing of meteorological Data

Invariant Laws - Each 15-minute average datum is examined and if its value has remained "constant" for 3 hours, then it is flagged. For each parameter, the limits of variation are:

a) wind speed $\pm$ 0.4 mps

b) wind direction $\pm$5degree range

c) differential temperature $\pm$ 0.2 ^{o}c

Sensor Thresholds:

DNPS = 0.9 mph → 0.4 mps
ZION = 0.7 " 0.3 "
LSCS = 0.7 " 0.3 "
BRWD = 0.8 " 0.4 "
BYRN = 0.8 " 0.4 "
SOQD = 0.8 " 0.4 "

4) All meteorological averages for any level are to be available on command. Any missing data should be represented by 999.

B. Downwind Directions

Wind Direction	Downwind Direction (Sector Name)	Adjacent Sectors
N	S (J)	H, K
NNE	SSW (K)	J, L
NE	SW (L)	K, M
ENE	WSW (M)	L, N
E	W (N)	M, P
ESE	WNW (P)	N, Q
SE	NW (Q)	P, R
SSE	NNW (R)	Q, A
S	N (A)	R, B
SSW	NNE (B)	A, C
SW	NE (C)	B, D
WSW	ENE (D)	C, E
W	E (E)	D, F
WNW	ESE (F)	E, G
NW	SE (G)	F, H
NNW	SSE (H)	G, J

TABLE 7.1-5

ATMOSPHERIC STABILITY CLASSES

DESCRIPTION	PASQUILL STABILITY CLASS	σ_θ (SEE NOTE, BELOW)	TEMPERATURE CHANGE WITH HEIGHT (°C/100 m)
Extremely Unstable	A	>22.4°	<1.9
Moderately Unstable	B	17.5 to 22.4	-1.9 to -1.7
Slightly Unstable	C	12.5 to 17.5	-1.7 to -1.5
Neutral	D	7.5 to 12.5	-1.5 to -0.5
Slightly Stable	E	3.7 to 7.5	-0.5 to 1.5
Moderately Stable	F	2.1 to 3.7	1.5 to 4.0
Extremely Stable	G	0.0 to 2.1	>4.0

NOTE: σ_θ is the standard deviation of horizontal wind direction fluctuation over a period of 15 minutes to 1 hour.

TABLE 9.2-1 (Cont'd)

0.8 MeV

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS) *

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.608-05	3.998-06	7.336-07	4.076-07	1.818-07	9.756-08
B	2.943-05	1.235-05	4.349-06	1.309-06	2.408-07	1.296-07
C	4.341-05	2.072-05	8.772-06	3.302-06	7.449-07	2.206-07
D	6.871-05	3.607-05	1.782-05	8.374-06	2.837-06	1.173-06
E	9.656-05	5.333-05	2.758-05	1.384-05	5.301-06	2.457-06
F	1.416-04	8.065-05	4.337-05	2.305-05	9.509-06	4.741-06
G	2.087-04	1.234-04	6.963-05	3.862-05	1.667-05	8.621-06

RELEASE HEIGHT = 100. (METERS) **

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	9.964-06	3.765-06	7.305-07	4.061-07	1.812-07	9.723-08
B	9.048-06	8.142-06	3.869-06	1.270-06	2.399-07	1.291-07
C	8.352-06	8.511-06	6.198-06	2.920-06	7.250-07	2.191-07
D	8.264-06	7.756-06	6.824-06	4.921-06	2.243-06	1.035-06
E	8.413-06	8.036-06	7.157-06	5.701-06	3.234-06	1.783-06
F	8.493-06	8.276-06	7.585-06	6.224-06	3.918-06	2.425-06
G	8.547-06	8.451-06	8.021-06	6.974-06	4.706-06	3.026-06

* By, Bd, Zion - All release points
D, QC - Reactor building vents

** D1, D2/3, QC Chimneys
LSCS Vent Stack

TABLE 9.2-6

SPLIT SIGMA CORRECTION FACTORS, $C(\sigma_\theta, \Delta T)$

STABILITY CLASS DETERMINED BY DIFFERENTIAL TEMPERATURE, ΔT*	STABILITY CLASS DETERMINED BY HORIZONTAL VARIATION, σ_θ*						
	A	B	C	D	E	F	G
A	1.0	1.330	1.751	2.487	3.497	5.067	7.605
B	0.752	1.0	1.317	1.870	2.630	3.810	5.719
C	0.571	0.759	1.0	1.420	1.997	2.893	4.343
D	0.402	0.535	0.704	1.0	1.406	2.037	3.058
E	0.286	0.380	0.501	0.711	1.0	1.449	2.175
F	0.197	0.263	0.346	0.491	0.690	1.0	1.501
G	0.132	0.175	0.230	0.327	0.460	0.666	1.0

*See Table 7.1-5 for classification scheme.

TABLE 9.4-2

RELATIVE CONCENTRATION (χ/Q) (sec/m^3)

		RANGE (meters)					
		402	804	1609	3218	8045	16090
Case 1.	Stability A	<u>4.6E-5</u>	1.6E-5	5.2E-6	2.4E-6	7.2E-7	2.8E-7
	Stability B	4.4E-5	2.6E-5	8.9E-6	3.5E-6	[illegible].0E-6	3.9E-7
	Stability C	2.5E-5	<u>4.2E-5</u>	1.8E-5	5.9E-6	1.5E-6	5.6E-7
	Stability D	5.5E-7	1.8E-5	<u>3.1E-5</u>	1.7E-5	5.2E-6	2.0E-6
	Stability E	3.4E-12	1.8E-6	2.2E-5	<u>2.6E-5</u>	1.0E-5	4.7E-6
	Stability F	9.6E-23	1.6E-9	2.4E-5	1.3E-5	<u>1.5E-5</u>	8.0E-6
	Stability G	4.8E-81	1.1E-17	1.3E-9	5.5E-7	5.3E-6	<u>7.1E-6</u>
Case 2.	(All Stabilities)	3.5E-4	2.1E-4	1.1E-4	5.3E-5	1.8E-5	7.0E-6

Note: Maximum χ/Q values for each range have been underscored.

Appendix B

ODCS Class B Model

DRAFT

9.0 REAL-TIME OFFSITE DOSE ASSESSMENT

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9.0 REAL-TIME OFFSITE DOSE ASSESSMENT

9.1 INTRODUCTION

Under certain circumstances, it may be desirable (or necessary) to obtain an offsite radiological dose assessment using the actual, real-time meteorological conditions. The doses (dose rates) of immediate interest in situations such as this are the whole body and skin dose due to exposure to the noble gases and the organ doses due to inhalation of radioiodines and other "particulates."

Personnel at Commonwealth Edison Company (CECo) nuclear power plants have access to real-time meteorology, not only for their own station, but to all other CECo nuclear plant sites. In addition, forecasted estimates of meteorological tower parameters also will be (are) available.

This section of the manual describes a method which enables a user to quickly obtain an accurate estimate of offsite radiological doses using the real-time (or forecasted) meteorology. Two models are considered. For use in times less than 8 hours after an accident, conditions at the centerline of a straight-line gaussian plume are considered. A meandering plume model (equivalent to uniform distribution over a 22½° sector) is assumed for times greater than 8 hours after an accident or other situations where this more realistic model should be considered. Further, special lake-breeze effects are considered for the Zion Station because of its location. Lake effects are discussed in Section 9.4.

9.2 NOBLE GASES

9.2.1 Gamma Whole Body Dose

9.2.1.1 Straight-Line Gaussian Plume Model

The relative concentration of material at any point in a straight-line, gaussian plume is given by Equation 3.115 in "Meteorology and Atomic Energy" (M&AE) (Reference 6.7), reproduced here as Equation 9.1.

$$\chi/Q = (2\pi\sigma_y\sigma_z u)^{-1} \exp\left(\frac{-y^2}{2\sigma_y^2}\right)\left\{\exp\left[\frac{-(z-h)^2}{2\sigma_z^2}\right] + \exp\left[\frac{-(z+h)^2}{2\sigma_z^2}\right]\right\} \tag{9.1}$$

χ/Q Relative Downwind Concentration (sec/m^3)

The relative downwind concentration at point (x, y, z)

x Downwind Distance (m)

The distance, parallel to the wind direction, from the release point to the point of interest.

y Transverse Distance (m)

The distance, perpendicular to the direction in which the wind is blowing, from the release point to the point of interest.

z Vertical Distance (m)

The distance above the ground to the point of interest.

σ_y Horizontal Dispersion Coefficient (m)

The horizontal dispersion coefficient for use in the atmospheric dispersion models. (See Table 7.1-7.)

σ_z Vertical Dispersion Coefficient (m)

The vertical dispersion coefficient for use in the atmospheric dispersion models. (See Table 7.1-7.)

u Wind Speed (m/sec)

The wind speed (in the x-direction).

h Release Height (m)

The height of the release above ground level.

Both σ_y and σ_z are functions of the downwind distance, x, and the atmospheric stability class.

The gamma dose rate from a point monoenergetic source is given by Equation 7.34 in M&AE, reproduced here as Equation 9.2.

$$D'_\gamma(E_\gamma,r) = 0.0404\ \mu_a\ q\ E_\gamma\ (1+k\mu r)\ \exp(-\mu r)/r^2 \qquad (9.2)$$

$D'_\gamma(E_\gamma,r)$ Gamma Dose Rate (rad/sec)

The tissue dose rate due to a point source in air. The source is monoenergetic with energy, E_γ, and a distance, r, away.

E_γ Gamma Ray Energy (MeV/dis)

The gamma ray energy released per disintegration.

r	Distance	(m)	
	The distance between the point of interest and the point source.		
μ_a	Energy Absorption Coefficient	(1/m)	
	The gamma ray ene y absorption coefficient for .		
q	Source Strength	(Ci)	
	The point source strength expressed in curies.		
μ	Total Absorption Coefficient	(1/m)	
	The gamma ray total absorption coefficient for air.		
k	Ratio		
	$k = (\mu - \mu_a)/\mu_a$		
0.0404	Constant		
	This constant reconciles the units of this equation. This constant includes the factor 1.11 for the ratio of electron density of tissue to that of air.		

Both μ and μ_a are functions of the gamma ray energy.

The differential unit source strength in the plume is given by Equation 9.3.

$$dq = Q(\chi/Q)\ dx\ dy\ dz \qquad (9.3)$$

dq	Differential Source Strength	(Ci)
	The source in a unit volume expressed in curies.	

Q	Release Rate The release rate of material.	(Ci/sec)
dx dy dz	Volume Element The differential volume element.	(m^3)

The differential dose rate at any point at ground level along the centerline of the plume (x_o, 0, 0) from a volume element at any other point (x, y, z) may then be obtained by combining Equations 9.1, 9.2, and 9.3.

$$d\,D'_\gamma(E_\gamma,r) = 0.0404\ \mu_a\ Q\ (X/Q)\ E_\gamma\ (1+k\mu r)\ \frac{\exp(-\mu r)}{r^2}\ dx\ dy\ dz \qquad (9.4)$$

The distance r is given by Equation 9.5.

$$r^2 = (x-x_o)^2 + y^2 + z^2 \qquad (9.5)$$

The total dose at (x_o, 0, 0) due to the entire plume is then obtained by integrating Equation 9.4 over the full extent of the plume; namely, over x from 0 to infinity; over y from minus to plus infinity; and over z from 0 to infinity. It is not possible to obtain a closed-form solution for the integrals of Equation 9.4; the values of the integrals must be obtained by using numerical techniques.

The total dose integrals of Equation 9.4 are a function of release height, downwind range, atmospheric stability class, and the gamma ray energy. As these evaluations are lengthy, a library of these integrals was calculated for two release heights (0 and 100 meters); six downwind ranges (400 to 16,090 meters); seven atmospheric stability classes; and a range of gamma ray energies. Then knowing the gamma ray spectrum associated with radiodecay of specific noble gas nuclides of interest, a dose kernel can be calculated for each nuclide. These dose kernels are defined, such that the total gamma dose rate at points of interest may be determined from Equation 9.6.

$$D'_\gamma(R) = \sum_i \frac{Q_i \exp(-\lambda_i R/3600u)}{u} K_i(h,R,S) \qquad (9.6)$$

$D'_\gamma(R)$	Gamma Dose Rate The ground level gamma dose rate at downwind distance, R, due to gamma emitting airborne radioactivity.	(mrad/hr)
R	Downwind Distance The downwind distance of interest. (The downwind distance was denoted as x_o in Equation 9.5.)	(m)
Q_i	Release Rate The release rate of nuclide i.	(μCi/sec)
λ_i	Radiodecay Constant The radioactive decay constant for nuclide i. See Table 7.1-9.	(1/hr)
3600	Conversion Constant Converts hours to seconds.	(sec/hr)
$K_i(h,R,S)$	Dose Kernel The finite cloud dose kernel, a function of release height (h); downwind range (R); stability class (S); and nuclide i released.	$\frac{(\text{mrad/hr})(\text{m/sec})}{\mu\text{Ci/sec}}$

The dose kernels have been evaluated for 15 noble gas nuclides and an 0.8 MeV pseudonuclide (to approximate a "gross" release) and are presented in Table 9.2-1.

If one obtains the wind speed, direction, and stability class from the meteorological tower, and has an estimate of the release rate Q_i, then with Equation 9.6 and the data of Table 9.2-1, an offsite dose rate at any downwind distance of interest may be quickly estimated.

Similarly, an offsite dose may be estimated from the following:

$$D_\gamma (R) = \sum_i A_i \frac{\exp(- \lambda_i R/3600u)}{u} \frac{K_i(h,R,S)}{3600} \qquad (9.7)$$

D_γ (R)	Gamma Dose The time integrated dose at downwind distance R.	(mrem)
A_i	Accumulative Release The accumulative release of nuclide i over the time period of interest.	(µCi)

9.2.1.2 Meandering Plume Model

The ground level dose rate due to a plume meandering within a 22½° angular sector may be evaluated in a manner similar to that described in the previous subsection.

It can be shown that if the term:

$$\int_0^1 \exp\left[-(y + \frac{R\theta}{2} - R\theta t)^2/2\sigma_y^2\right] dt$$

is substituted for the term:

$$\exp(-y^2/2\sigma_y^2)$$

in Equation 9.1, then the revised Equations 9.1 and 9.4 will describe the dose rate due to the meandering plume. The term θ in the above expression is the angular sector width over which the plume is assumed to meander. Again, the total rate from the entire plume is obtained by numerical integration techniques and kernels for use in the dose rate and dose Equations 9.6 and 9.7 have been determined. The "K" kernels have been determined for 2 release heights (0 and 100 meters); 6 downwind ranges (400 to 16,090 meters); 7 atmospheric stability classes; 15 noble gas nuclides; and 1 pseudonuclide (0.8 MeV gamma ray, 34 minute half-life); these data are given in Table 9.2-2. If one obtains the wind, speed, direction, and stability class from the meteorological tower, and has an estimate of the release rate Q_i, then with Equation 9.6 and the data of Table 9.2-2, an offsite dose rate at any downwind distance of interest may be quickly estimated. And similarly, using Equation 9.7, an offsite dose may be determined.

9.2.1.3 Population Doses

The whole body population dose in downwind sectors to a distance of 10 miles from the station may also be of relevance in emergency planning considerations. Such a sector population may be calculated as follows:

$$D^{*}_{\gamma}(\theta) = \sum_{j=1}^{5} P_j(\theta) \left[D_{\gamma}(R_j) \times D_{\gamma}(R_{j+1}) \right]^{1/2} \Big/ 1000 \qquad (9.8)$$

$D^{*}_{\gamma}(\theta)$ Whole Body Population Dose (man-rem)
The gamma whole body dose to the population in angular sector θ.

$P_j(\theta)$ Population (persons)
The population in radial sector j and angular sector θ. See Table 9.2-3.

$D_\gamma(R_j)$ Gamma Dose (mrad)

The gamma dose to an individual at downwind distance R_j. (From Equation 9.7.)

R_j Radial Sector Boundary Distance (m)

The distance to inner radial boundary of the population sector. The values of R_j are the same as used in Tables 9.2-1, 9.2-2, 9.2-4, and 9.2-5; namely, 400, 800, 1609, 3218, and 8045 meters, respectively. The distance to the outermost sector is 16,090 meters. No population is presumed to be present in the 0-400 meter sector.

1000 Conversion Constant (mrem/rem)

Converts mrem to rem.

The term $\left[D_\gamma(R_j) \times D_\gamma(R_{j+1}) \right]^{1/2}$ in Equation 9.8 is the geometric mean dose in the angular sector bounded by R_j and R_{j+1}.

9.2.2 Skin Dose

The dose rate to the skin (see Subsection 2.1.1) is due to two components; gamma rays and beta rays. The gamma component is determined from Equation 9.6; the beta component is determined from the following equation:

$$D'_\beta (R) = \frac{1}{8760} \sum_i \bar{L}_i \left[\left(\chi/Q \right)_s Q'_{is} + \left(\chi/Q \right)_g Q'_{ig} \right] \qquad (9.9)$$

D'_β (R) Skin Dose Rate (mrem/hr)

The dose rate at (R) to the skin due to beta-emitting airborne radioactivity.

8760 Conversion Constant (hr/yr)
Converts years to hours.

$\overline{L}_i$ Beta Skin Dose Constant (mrem/yr/μCi/m^3)
The skin dose factor due to beta emissions for each identified noble gas radionuclide i. This factor accounts for the attenuation of beta radiation during passage through 7 mg/cm^2 of dead skin. Values for specific nuclides are given in Table 7.1-13.

$(\chi/Q)_s$ Relative Effluent Concentration, Stack Release (sec/m^3)
The relative effluent concentration at ground level due to stack releases.

$(\chi/Q)_g$ Relative Effluent Concentration, Ground Level Release (sec/m^3)
The relative effluent concentration at ground level due to ground level releases.

Q'_{is} Release Rate From Stack, Adjusted for Radiodecay (μCi/sec)
The release rate for radionuclide i from a stack adjusted for radiodecay in transit.

$$Q'_{is} = Q_{is} \times \exp(-\lambda_i R/3600u_s) \quad (9.10)$$

Q_{is} Release Rate From Stack (μCi/sec)
The release rate of radionuclide i from the stack.

u_s Wind Speed, Stack Elevation (m/sec)
The wind speed at the elevation of the top of the stack.

Q'_{ig} Release Rate, Ground Level, Adjusted for Radiodecay (μCi/sec)
The release rate for radionuclide i at ground level, adjusted for radiodecay in transit.

$$Q'_{ig} = Q_{ig} \times \exp(-\lambda_i R/3600 u_g) \quad (9.11)$$

Q_{ig} Release Rate at Ground Level (μCi/sec)
The release rate of radionuclide i at ground level.

u_g Wind Speed, Ground Level (m/sec)
The wind speed at the lowest position on the meteorological tower.

The relative concentration, χ/Q, is a function of downwind range, atmospheric stability class, release height, meander model, and wind speed. To preclude extensive table look-ups, a table of the parameter $(u\chi/Q)$ has been prepared for the same two release heights (0 and 100 meters): six downwind ranges (400 to 16,090 meters); and seven atmospheric stability classes used to develop the whole body dose kernels.

9.2.2.1 χ/Q Modeling

Two models are considered for χ/Q evaluations. The first is the straight-line gaussian plume model considered for times less than 8 hours after an accident. The ground level relative concentration for a plume directly overhead is given by the following equations:

$$u_s \left(\chi/Q\right)_s = \frac{1}{\pi\sigma_y\sigma_z} \exp -\left(\frac{h^2}{2\sigma_z^2}\right) \quad (9.12)$$

$$u_g \left(\chi/Q\right)_g = \frac{1}{\pi S_y S_z} \quad (9.13)$$

σ_y Horizontal Dispersion Coefficient (m)

The horizontal dispersion coefficient for use in the atmospheric dispersion models. (See Table 7.1-7.)

σ_z Vertical Dispersion Coefficient (m)

The vertical dispersion coefficient for use in the atmospheric dispersion models. (See Table 7.1-7.)

h Effluent Release Height (m)

The height above grade at which the effluent is effectively released.

S_y Corrected Horizontal Dispersion Coefficient (m)

The horizontal dispersion coefficient corrected for building wake effects.

$$S_y = \left(\sigma_y^2 + A/_{2\pi}\right)^{\frac{1}{2}} \tag{9.14}$$

A Building Cross-Sectional Area

The effective building cross-sectional area determining the downwind wake effect.

S_z Corrected Vertical Dispersion Coefficient (m)

The vertical dispersion coefficient corrected for building wake effects.

$$S_z = \left(\sigma_z^2 + A/_{2\pi}\right)^{\frac{1}{2}} \tag{9.15}$$

The values of S_y and S_z are further limited by the condition:

$$S_y \leq \sqrt{3\sigma_y} \tag{9.16}$$

$$S_z \leq \sqrt{3\sigma_z} \tag{9.17}$$

The second model is for the case of a meandering plume to be used for times greater than 8 hours after an accident or other times when deemed appropriate. The ground level relative concentrations at the centerline of the sector are given by the following equations:

$$u_s \left(\chi/Q\right)_s = \frac{2.032}{R\sigma_z} \exp -\left(\frac{h^2}{2\sigma_z^2}\right) \tag{9.18}$$

$$u_g \left(\chi/Q\right)_g = \frac{2.032}{R\ S_z} \tag{9.19}$$

The values of $(u\chi/Q)$ for the straight-line gaussian plume are given in Table 9.2-4. The values of $(u\chi/Q)$ for the meandering plume are given in Table 9.2-5.

If one obtains the wind speed, direction, and stability class from the meteorological tower, the downwind χ/Q may be calculated as follows:

$$\left(\chi/Q\right)_s = \frac{1}{u_s} \left(u\chi/Q\right)_s \tag{9.20}$$

$$\left(\chi/Q\right)_g = \frac{1}{u_g} \left(u\chi/Q\right)_g \tag{9.21}$$

Values of $u\chi/Q$ are taken from Tables 9.2-4 or 9.2-5; the wind speed u_s or u_g is from the upper or lower level of the meteorological tower as is appropriate for the release.

The formulations of $(u\chi/Q)$ have been heretofore developed assuming that a single stability class determination is applicable for describing both horizontal and vertical dispersion. And this formulation is consistent with data available in historical meteorological data files. However, when real-time meteorological information is available, data to support a more accurate straight-line gaussian plume modeling scheme is available.

The vertical differential temperature measurements, ΔT, (indicative of dispersion in the vertical direction) will be used to determine the "vertical stability class" which, in turn, determines σ_z. The vane directional variation measurement, σ_θ, is more indicative of horizontal dispersion; therefore, σ_θ will be used to determine a "horizontal stability class" which, in turn, determines σ_y. Rather than developing tables of all possible combinations of σ_y and σ_z, the numerical effect of these differing atmospheric conditions is consolidated into a single correction term. The corrected $(u\chi/Q)$ is determined as follows:

$$(u\chi/Q)_{corrected} = C(\sigma_\theta, \Delta T) \times (u\chi/Q) \qquad (9.22)$$

$(u\chi/Q)$ Normalized Dispersion Factor $(1/m^2)$

The value of $(u\chi/Q)$ from Table 9.2-4 for the distance of interest and stability class as determined from the meteorological tower differential temperature measurements.

$(u\chi/Q)_{corrected}$ Corrected Normalized Dispersion Factor $(1/m^2)$

The value of $(u\chi/Q)$ corrected for differing horizontal and vertical stability classes.

$C(\sigma_\theta, \Delta T)$ Correction Factor

The correction factor to account for a difference in stability class as determined by the ΔT and σ_θ methods. See Table 9.2-6.

These correction factors were readily derivable for use in $(u\chi/Q)$ formulation. However, an equivalent correction factor for use in the finite cloud models (Subsection 9.2.1) is not easily derivable. But because of the poor gamma attenuating properties of air, the effect of modest additional dispersion on the cloud would have little effect on the final value of the kernel. Hence a correction for the gamma dose kernel will not be pursued further.

9.2.2.2 Skin Dose Assessment

Combining Equation 9.9 with Equations 9.10, 9.11, 9.20, and 9.21 (and 9.22, if applicable) results in the final expression for beta skin dose rate:

$$D'_\beta\ (R) = \frac{1}{8760} \sum_i \bar{L}_i \left[\frac{1}{u_s} \left(u\chi/_Q \right)_s Q_{is} \exp(-\lambda_i R/3600u_s) + \frac{1}{u_g} \left(u\chi/_Q \right)_g Q_{ig} \exp(-\lambda_i R/3600u_g) \right] \quad (9.23)$$

The total skin dose rate is obtained by adding the gamma component from Equation 9.6 and the beta component from Equation 9.23:

$$D'_{skin}(R) = D'_\gamma\ (R) + D'_\beta\ (R) \quad (9.24)$$

D'_{skin} (R) Skin Dose Rate (mrem/hr)

The dose rate, at downwind distance R, to the skin due to beta and gamma rays.

A beta skin dose may be obtained from the following equation (which follows from Equation 9.23):

$$D_\beta (R) = \frac{1}{8760 \times 3600} \sum_i \bar{L}_i \left[\frac{1}{u_s} \left(u\chi/Q \right)_s A_{is} \exp(-\lambda_i R/3600 u_s) + \frac{1}{u_g} \left(u\chi/Q \right)_g A_{ig} \exp(-\lambda_i R/3600 u_g) \right] \tag{9.25}$$

D_β (R)	Beta Skin Dose	(mrem)
	The dose at downwind distance R to the skin due to beta rays.	

A total skin dose may be obtained by combining the results of Equations 9.7 and 9.25.

$$D_{skin}(R) = D_\gamma (R) + D_\beta (R) \tag{9.26}$$

$D_{skin}(R)$	Skin Dose	(mrem)
	The skin dose at (R) due to gamma and beta rays emitted from airborne radionuclides.	

9.2.3 Symbols Used in Section 9.2

SYMBOL	NAME	UNIT
χ/Q	Relative Downwind Concentration	(sec/m^3)
x	Downwind Distance	(m)
y	Transverse Distance	(m)
z	Vertical Distance	(m)
σ_y	Horizontal Dispersion Coefficient	(m)
σ_z	Vertical Dispersion Coefficient	(m)
u	Wind Speed	(m/sec)
h	Effluent Release Height	(m)
$D'_\gamma(E_\gamma, r)$	Gamma Dose Rate	(rad/sec)
E_γ	Gamma Ray Energy	(MeV/dis)
r	Distance	(m)
μ_a	Energy Absorption Coefficient	(1/m)
q	Source Strength	(Ci)
k	Ratio	
μ	Total Absorption Coefficient	(1/m)
dq	Differential Source Strength	(Ci)
Q	Release Rate	(Ci/sec)
$dx\ dy\ dz$	Volume Element	(m^3)
$D'_\gamma(R)$	Gamma Dose Rate	(mrad/hr)
R	Downwind Distance	(m)
Q_i	Release Rate	(μCi/sec)
λ_i	Radiodecay Constant	(1/hr)

SYMBOL	NAME	UNIT
$K_i\ (h,R,S)$	Dose Kernel	$\frac{(mrad/hr)(m/sec)}{\mu Ci/sec}$
$D_\gamma(R)$	Gamma Dose	(mrad)
A_i	Accumulative Release	(μCi)
θ	Angular Sector	
$D^*_\gamma(\theta)$	Whole Body Population Dose	(man-rem)
$P_j(\theta)$	Population	(persons)
$D_\gamma(R_j)$	Gamma Dose	(mrad)
R_j	Radial Sector Boundary Distance	(m)
$D'_\beta\ (R)$	Beta Skin Dose Rate	(mrem/hr)
$\overline{L}_i$	Beta Skin Dose Constant	(mrem/yr per $\mu Ci/m^3$)
$(\chi/Q)_s$	Relative Effluent Concentration, Stack Release	(sec/m^3)
Q'_{is}	Release Rate From Stack, Adjusted for Radiodecay	(μCi/sec)
Q_{is}	Release Rate From Stack	(μCi/sec)
u_s	Wind Speed, Stack Elevation	(m/sec)
$(\chi/Q)_g$	Relative Effluent Concentration, Ground Level	(sec/m^3)
Q'_{ig}	Release Rate, Ground Level, Adjusted for Radiodecay	(μCi/sec)
Q_{ig}	Release Rate at Ground Level	(μCi/sec)
u_g	Wind Speed, Ground Level	(m/sec)
S_y	Corrected Horizontal Dispersion Coefficient	(m)
A	Building Cross-Sectional Area	(m^2)
S_z	Corrected Vertical Dispersion Coefficient	(m)
ΔT	Vertical Differential Temperature	(°C)

SYMBOL	NAME	UNIT
$(u\chi/Q)$	Normalized Dispersion Factor	$(1/m^2)$
$(u\chi/Q)_{corrected}$	Corrected Normalized Dispersion Factor	$(1/m^2)$
$C(\sigma_\theta,\Delta T)$	Correction Factor	
$(u\chi/Q)_s$	Dispersion Factor, Stack Release	$(1/m^2)$
$(u\chi/Q)_g$	Dispersion Factor, Ground Release	$(1/m^2)$
$D'_{skin}(R)$	Skin Dose Rate	(mrem/hr)
$D_\beta(R)$	Beta Skin Dose	(mrem)
$D_{skin}(R)$	Skin Dose	(mrem)

9.2.4 Constants Used in Section 9.2

NUMERICAL VALUE	NAME	UNIT
0.0404	Constant	
3600	Conversion Constant	(sec/hr)
1000	Conversion Constant	(mrem/rem)
8760	Conversion Constant	(hr/yr)
2.032	Constant	

TABLE 9.2-1

FINITE CLOUD GAMMA TISSUE DOSE KERNEL

STRAIGHT-LINE GAUSSIAN PLUME MODEL

Kr-83m

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	9.960-08	1.514-08	2.368-09	1.266-09	5.536-10	2.961-10
B	2.711-07	6.870-08	1.707-08	4.291-09	7.362-10	3.937-10
C	5.435-07	1.557-07	4.396-08	1.251-08	2.374-09	6.753-10
D	1.311-06	4.127-07	1.337-07	4.600-08	1.171-08	4.239-09
E	2.544-06	8.341-07	2.723-07	9.754-08	2.733-08	1.085-08
F	5.628-06	1.849-06	6.031-07	2.223-07	6.575-08	2.740-08
G	1.287-05	4.414-06	1.476-06	5.491-07	1.634-07	6.825-08

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	4.864-08	1.430-08	2.358-09	1.261-09	5.513-10	2.948-10
B	1.344-08	3.503-08	1.475-08	4.156-09	7.330-10	3.920-10
C	1.634-10	2.111-08	2.[illegible]-08	1.068-08	2.305-09	6.700-10
D	6.950-11	1.323-10	1.017-08	1.584-08	8.162-09	3.595-09
E	6.601-11	7.028-11	1.283-09	8.634-09	1.010-08	6.269-09
F	6.415-11	6.331-11	6.503-11	6.078-10	4.226-09	5.168-09
G	6.389-11	6.257-11	5.870-11	5.032-11	4.551-11	7.098-10

TABLE 9.2-1 (Cont'd)

Kr-85m

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	4.216-06	8.999-07	1.515-07	8.268-08	3.644-08	1.951-08
B	8.016-06	3.142-06	9.953-07	2.738-07	4.837-08	2.593-08
C	1.192-05	5.515-06	2.178-06	7.425-07	1.534-07	4.435-08
D	1.878-05	9.843-06	4.719-06	2.111-06	6.561-07	2.570-07
E	2.823-05	1.458-05	7.460-06	3.638-06	1.304-06	5.784-07
F	3.824-05	2.191-05	1.183-05	6.195-06	2.474-06	1.187-06
G	5.654-05	3.335-05	1.889-05	1.048-05	4.495-06	2.262-06

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	2.468-06	8.471-07	1.511-07	8.245-08	3.634-08	1.946-08
B	2.145-06	1.952-06	8.746-07	2.658-07	4.824-08	2.587-08
C	1.916-06	2.000-06	1.455-06	6.470-07	1.492-07	4.408-08
D	1.861-06	1.734-06	1.533-06	1.113-06	4.972-07	2.230-07
E	1.896-06	1.790-06	1.578-06	1.252-06	7.131-07	3.929-07
F	1.914-06	1.846-06	1.664-06	1.329-06	8.199-07	5.128-07
G	1.928-06	1.891-06	1.772-06	1.492-06	9.489-07	5.974-07

TABLE 9.2-1 (Cont'd)

Kr-85

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	4.893-08	1.159-08	2.062-09	1.138-09	5.051-10	2.708-10
B	9.031-08	3.723-08	1.272-08	3.700-09	6.694-10	3.598-10
C	1.333-07	6.325-08	2.622-08	9.592-09	2.092-09	6.137-10
D	2.104-07	1.106-07	5.425-08	2.521-08	8.301-09	3.372-09
E	2.951-07	1.634-07	8.439-08	4.203-08	1.582-08	7.228-09
F	4.314-07	2.465-07	1.328-07	7.037-08	2.888-08	1.422-08
G	6.348-07	3.764-07	2.126-07	1.181-07	5.087-08	2.623-08

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	400.	DOWNWIND DISTANCE(METERS) 800.	1609.	3218.	8045.	16090.
A	2.983-08	1.091-08	2.054-09	1.134-09	5.036-10	2.700-10
B	2.691-08	2.413-08	1.127-08	3.591-09	6.676-10	3.587-10
C	2.470-08	2.519-08	1.823-08	8.439-09	2.036-09	6.097-10
D	2.437-08	2.277-08	1.995-08	1.435-08	6.475-09	2.958-09
E	2.482-08	2.361-08	2.091-08	1.651-08	9.346-09	5.140-09
F	2.507-08	2.435-08	2.219-08	1.802-08	1.121-08	6.927-09
G	2.523-08	2.489-08	2.353-08	2.026-08	1.340-08	8.526-09

TABLE 9.2-1 (Cont'd)

Kr-87

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.376-05	3.534-06	6.898-07	3.868-07	1.752-07	9.433-08
B	2.509-05	1.064-05	3.854-06	1.206-06	2.313-07	1.252-07
C	3.696-05	1.772-05	7.626-06	2.930-06	6.961-07	2.118-07
D	5.854-05	3.073-05	1.528-05	7.270-06	2.505-06	1.059-06
E	8.247-05	4.542-05	2.354-05	1.193-05	4.596-06	2.163-06
F	1.214-04	6.877-05	3.693-05	1.971-05	8.194-06	4.091-06
G	1.799-04	1.057-04	5.930-05	3.290-05	1.435-05	7.353-06

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	8.584-06	3.340-06	6.865-07	3.853-07	1.745-07	9.400-08
B	7.809-06	7.076-06	3.444-06	1.172-06	2.304-07	1.247-07
C	7.215-06	7.385-06	5.441-06	2.606-06	6.783-07	2.103-07
D	7.135-06	6.737-06	5.966-06	4.346-06	2.007-06	9.414-07
E	7.260-06	6.968-06	6.240-06	5.016-06	2.875-06	1.600-06
F	7.327-06	7.169-06	6.593-06	5.454-06	3.491-06	2.177-06
G	7.372-06	7.315-06	6.959-06	6.079-06	4.170-06	2.725-06

TABLE 9.2-1 (Cont'd)

Kr-88

Kernel Units Are (mrad/hr)(m/sec)/(µCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	3.115-05	8.313-06	1.678-06	9.466-07	4.311-07	2.324-07
B	5.643-05	2.427-05	9.040-06	2.911-06	5.685-07	3.082-07
C	8.319-05	4.003-05	1.754-05	6.895-06	1.693-06	5.205-07
D	1.324-04	6.921-05	3.460-05	1.665-05	5.866-06	2.519-06
E	1.876-04	1.025-04	5.308-05	2.709-05	1.056-05	5.045-06
F	2.785-04	1.561-04	8.328-05	4.452-05	1.861-05	9.375-06
G	4.169-04	2.420-04	1.344-04	7.426-05	3.248-05	1.660-05

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.967-05	7.873-06	1.670-06	9.427-07	4.294-07	2.315-07
B	1.796-05	1.636-05	8.108-06	2.830-06	5.663-07	3.070-07
C	1.664-05	1.707-05	1.268-05	6.161-06	1.650-06	5.167-07
D	1.649-05	1.564-05	1.392-05	1.019-05	4.754-06	2.252-06
E	1.677-05	1.616-05	1.455-05	1.178-05	6.791-06	3.797-06
F	1.692-05	1.661-05	1.534-05	1.280-05	8.297-06	5.191-06
G	1.702-05	1.693-05	1.616-05	1.422-05	9.914-06	6.549-06

TABLE 9.2-1 (Cont'd)

Kr-89

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	2.969-05	7.689-06	1.507-06	8.457-07	3.832-07	2.064-07
B	5.402-05	2.299-05	8.376-06	2.632-06	5.059-07	2.738-07
C	7.958-05	3.821-05	1.650-05	6.377-06	1.521-06	4.632-07
D	1.261-04	6.620-05	3.296-05	1.572-05	5.442-06	2.306-06
E	1.777-04	9.783-05	5.073-05	2.573-05	9.953-06	4.695-06
F	2.617-04	1.482-04	7.957-05	4.248-05	1.769-05	8.851-06
G	3.878-04	2.279-04	1.278-04	7.090-05	3.091-05	1.587-05

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.858-05	7.267-06	1.500-06	8.423-07	3.818-07	2.055-07
B	1.693-05	1.534-05	7.489-06	2.558-06	5.040-07	2.728-07
C	1.566-05	1.602-05	1.181-05	5.676-06	1.482-06	4.599-07
D	1.550-05	1.465-05	1.297-05	9.451-06	4.369-06	2.053-06
E	1.577-05	1.515-05	1.358-05	1.092-05	6.262-06	3.485-06
F	1.591-05	1.558-05	1.434-05	1.188-05	7.619-06	4.751-06
G	1.601-05	1.590-05	1.513-05	1.324-05	9.110-06	5.962-06

TABLE 9.2-1 (Cont'd)

Kr-90

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	4.079-05	1.038-05	2.001-06	1.120-06	5.057-07	2.721-07
B	7 [illegible]6-05	3.147-05	1.131-05	3.512-06	6.681-07	3.611-07
C	1.100-04	5.258-05	2.250-05	8.596-06	2.023-06	6.119-07
D	1.744-04	9.141-05	4.531-05	2.146-05	7.363-06	3.090-06
E	2.458-04	1.352-04	6.995-05	3.532-05	1.356-05	6.364-06
F	3.619-04	2.049-04	1.099-04	5.853-05	2.424-05	1.210-05
G	5.365-04	3.151-04	1.767-04	9.787-05	4.254-05	2.180-05

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	2.537-05	9.805-06	1.992-06	1.116-06	5.039-07	2.711-07
B	2.301-05	2.087-05	1.010-05	3.412-06	6.656-07	3.597-07
C	2.122-05	2.175-05	1.601-05	7.637-06	1.970-06	6.075-07
D	2.097-05	1.979-05	1.753-05	1.276-05	5.881-06	2.749-06
E	2.133-05	2.046-05	1.832-05	1.472-05	8.430-06	4.686-06
F	2.153-05	2.105-05	1.935-05	1.599-05	1.021-05	6.368-06
G	2.166-05	2.148-05	2.042-05	1.7[illegible]2-05	1.219-05	7.949-06

TABLE 9.2-1 (Cont'd)

Xe-131m

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	7.141-07	1.208-07	1.927-08	1.036-08	4.541-09	2.429-09
B	1.653-06	5.043-07	1.354-07	3.490-08	6.035-09	3.230-09
C	2.822-06	1.031-06	3.312-07	9.974-08	1.938-08	5.535-09
D	5.280-06	2.226-06	8.765-07	3.360-07	9.163-08	3.399-08
E	8.320-06	3.758-06	1.575-06	6.530-07	2.021-07	8.346-08
F	1.379-05	6.548-06	2.890-06	1.286-06	4.384-07	1.944-07
G	2.296-05	1.152-05	5.408-06	2.557-06	9.258-07	4.246-07

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	3.699-07	1.141-07	1.921-08	1.033-08	4.527-09	2.422-09
B	2.073-07	2.766-07	1.177-07	3.385-08	6.017-09	3.220-09
C	1.183-07	2.250-07	2.000-07	8.573-08	1.884-08	5.499-09
D	9.423-08	1.022-07	1.433-07	1.377-07	6.569-08	2.903-08
E	9.314-08	9.197-08	1.001-07	1.165-07	8.742-08	5.110-08
F	9.233-08	8.978-08	8.450-08	7.702-08	6.818-08	5.364-08
G	9.251-08	9.053-08	8.446-08	7.071-08	4.778-08	3.609-08

TABLE 9.2-1 (Cont'd)

Xe-133m

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.268-06	2.415-07	3.977-08	2.157-08	9.483-09	5.076-09
B	2.705-06	9.182-07	2.687-07	7.191-08	1.260-08	6.748-09
C	4.392-06	1.755-06	6.195-07	1.994-07	4.014-08	1.155-08
D	7.774-06	3.520-06	1.496-06	6.151-07	1.793-07	6.855-08
E	1.183-05	5.686-06	2.555-06	1.131-06	3.750-07	1.605-07
F	1.894-05	9.471-06	4.451-06	2.100-06	7.646-07	3.514-07
G	3.058-05	1.601-05	7.920-06	3.941-06	1.517-06	7.228-07

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	6.965-07	2.276-07	3.965-08	2.150-08	9.457-09	5.062-09
B	4.922-07	5.359-07	2.350-07	6.978-08	1.256-08	6.728-09
C	3.735-07	4.913-07	3.941-07	1.727-07	3.904-08	1.148-08
D	3.431-07	3.343-07	3.486-07	2.869-07	1.325-07	5.908-08
E	3.467-07	3.318-07	3.120-07	2.849-07	1.833-07	1.039-07
F	3.483-07	3.371-07	3.081-07	2.564-07	1.792-07	1.230-07
G	3.503-07	3.438-07	3.223-07	2.722-07	1.771-07	1.177-07

TABLE 9.2-1 (Cont'd)

Xe-133

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.463-06	2.707-07	4.385-08	2.370-08	1.040-08	5.564-09
B	3.091-06	1.056-06	3.019-07	7.940-08	1.382-08	7.396-09
C	4.935-06	2.016-06	7.085-07	2.232-07	4.420-08	1.267-08
D	8.463-06	3.974-06	1.717-06	7.036-07	2.021-07	7.643-08
E	1.253-05	6.289-06	2.903-06	1.297-06	4.274-07	1.818-07
F	1.937-05	1.016-05	4.961-06	2.389-06	8.737-07	4.008-07
G	3.018-05	1.654-05	8.560-06	4.382-06	1.721-06	8.223-07

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	7.964-07	2.553-07	4.373-08	2.364-08	1.037-08	5.549-09
B	5.714-07	6.105-07	2.633-07	7.706-08	1.378-08	7.378-09
C	4.357-07	5.629-07	4.460-07	1.928-07	4.300-08	1.259-08
D	3.954-07	3.848-07	3.963-07	3.225-07	1.480-07	6.562-08
E	3.993-07	3.806-07	3.561-07	3.209-07	2.048-07	1.160-07
F	4.010-07	3.864-07	3.505-07	2.880-07	1.989-07	1.364-07
G	4.034-07	3.945-07	3.669-07	3.044-07	1.932-07	1.280-07

TABLE 9.2-1 (Cont'd)

Xe-135m

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	9.576-06	2.263-06	4.026-07	2.222-07	9.863-08	5.288-08
B	1.775-05	7.280-06	2.483-06	7.224-07	1.307-07	7.026-08
C	2.629-05	1.240-05	5.123-06	1.873-06	4.085-07	1.198-07
D	4.171-05	2.178-05	1.064-05	4.929-06	1.621-06	6.584-07
E	5.873-05	3.230-05	1.659-05	8.235-06	3.093-06	1.412-06
F	8.632-05	4.895-05	2.621-05	1.383-05	5.656-06	2.781-06
G	1.277-04	7.517-05	4.218-05	2.330-05	9.994-06	5.143-06

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	5.827-06	2.130-06	4.011-07	2.215-07	9.835-08	5.272-08
B	5.229-06	4.712-06	2.199-06	7.011-07	1.304-07	7.004-08
C	4.785-06	4.903-06	3.558-06	1.648-06	3.975-07	1.191-07
D	4.716-06	4.412-06	3.879-06	2.799-06	1.264-06	5.776-07
E	4.803-06	4.571-06	4.054-06	3.217-06	1.823-06	1.003-06
F	4.850-06	4.712-06	4.296-06	3.493-06	2.179-06	1.349-06
G	4.882-06	4.816-06	4.554-06	3.924-06	2.538-06	1.655-06

TABLE 9.2-1 (Cont'd)

Xe-135

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	6.212-06	1.376-06	2.352-07	1.287-07	5.683-08	3.044-08
B	1.166-05	4.658-06	1.518-06	4.244-07	7.543-08	4.046-08
C	1.726-05	8.082-06	3.255-06	1.136-06	2.383-07	6.915-08
D	2.713-05	1.428-05	6.907-06	3.144-06	9.964-07	3.948-07
E	3.789-05	2.109-05	1.086-05	5.348-06	1.952-06	8.743-07
F	5.503-05	3.167-05	1.713-05	9.023-06	3.651-06	1.767-06
G	8.047-05	4.807-05	2.731-05	1.520-05	6.557-06	3.328-06

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	3.690-06	1.294-06	2.344-07	1.283-07	5.668-08	3.036-08
B	3.269-06	2.941-06	1.337-06	4.118-07	7.522-08	4.035-08
C	2.959-06	3.046-06	2.206-06	9.928-07	2.319-07	6.872-08
D	2.896-06	2.697-06	2.365-06	1.706-06	7.624-07	3.437-07
E	2.952-06	2.792-06	2.461-06	1.942-06	1.098-06	6.033-07
F	2.981-06	2.881-06	2.606-06	2.090-06	1.287-06	7.988-07
G	3.002-06	2.950-06	2.773-06	2.353-06	1.515-06	9.541-07

TABLE 9.2-1 (Cont'd)

Xe-137

Kernel Units Are (mrad/hr)(m/sec)/(µCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	4.191-06	1.003-06	1.815-07	1.005-07	4.478-08	2.403-08
B	7.729-06	3.193-06	1.099-06	3.240-07	5.931-08	3.192-08
C	1.141-05	5.416-06	2.256-06	8.300-07	1.840-07	5.435-08
D	1.801-05	9.465-06	4.648-06	2.166-06	7.176-07	2.933-07
E	2.527-05	1.399-05	7.226-06	3.607-06	1.360-06	6.241-07
F	3.698-05	2.110-05	1.137-05	6.028-06	2.476-06	1.221-06
G	5.445-05	3.225-05	1.820-05	1.011-05	4.367-06	2.244-06

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	2.560-06	9.449-07	1.808-07	1.001-07	4.464-08	2.395-08
B	2.309-06	2.076-06	9.752-07	3.145-07	5.913-08	3.181-08
C	2.119-06	2.165-06	1.573-06	7.316-07	1.791-07	5.399-08
D	2.090-06	1.957-06	1.719-06	1.240-06	5.620-07	2.579-07
E	2.128-06	2.027-06	1.799-06	1.428-06	8.095-07	4.463-07
F	2.149-06	2.090-06	1.907-06	1.553-06	9.714-07	6.019-07
G	2.163-06	2.136-06	2.021-06	1.743-06	1.159-06	7.415-07

TABLE 9.2-1 (Cont'd)

Xe-138

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0 (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.936-05	4.978-06	9.696-C7	5.441-07	2.459-07	1.323-07
B	3.532-05	1.496-05	5.426-06	1.699-06	3.248-07	1.756-07
C	5.211-05	2.494-05	1.074-05	4.121-06	9.808-07	2.975-07
D	8.274-05	4.329-05	2.150-05	1.023-05	3.530-06	1.491-06
E	1.169-04	6.409-05	3.315-05	1.679-05	6.460-06	3.047-06
F	1.726-04	9.731-05	5.206-05	2.774-05	1.151-05	5.764-06
G	2.569-04	1.502-04	8.381-05	4.636-05	2.021-05	1.031-05

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.208-05	4.708-06	9.650-07	5.419-07	2.450-07	1.319-07
B	1.098-05	9.959-06	4.849-06	1.651-06	3.236-07	1.750-07
C	1.014-05	1.039-05	7.664-06	3.666-06	9.556-07	2.954-07
D	1.002-05	9.466-06	8.393-06	6.119-06	2.828-06	1.326-06
E	1.020-05	9.786-06	8.772-06	7.061-06	4.048-06	2.255-06
F	1.029-05	1.007-05	9.262-06	7.670-06	4.916-06	3.068-06
G	1.035-05	1.027-05	9.773-06	8.543-06	5.869-06	3.839-06

TABLE 9.2-1 (Cont'd)

Ar-41

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	2.273-05	5.898-06	1.141-06	6.398-07	2.885-07	1.552-07
B	4.131-05	1.760-05	6.411-06	2.007-06	3.812-07	2.060-07
C	6.090-05	2.922-05	1.253-05	4.879-06	1.156-06	3.491-07
D	9.673-05	5.063-05	2.520-05	1.201-05	4.170-06	1.762-06
E	1.366-04	7.494-05	3.879-05	1.967-05	7.613-06	3.596-06
F	2.017-04	1.138-04	6.090-05	3.249-05	1.350-05	6.788-06
G	2.997-04	1.755-04	9.810-05	5.427-05	2.360-05	1.212-05

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.425-05	5.573-06	1.136-06	6.373-07	2.874-07	1.546-07
B	1.300-05	1.177-05	5.731-06	1.949-06	3.798-07	2.051-07
C	1.204-05	1.230-05	9.052-06	4.341-06	1.126-06	3.467-07
D	1.193-05	1.126-05	9.966-06	7.244-06	3.345-06	1.567-06
E	1.213-05	1.165-05	1.044-05	8.390-06	4.798-06	2.666-06
F	1.224-05	1.198-05	1.103-05	9.146-06	5.851-06	3.642-06
G	1.232-05	1.222-05	1.164-05	1.020-05	7.016-06	4.580-06

TABLE 9.2-1 (Cont'd)

0.8 MeV

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.608-05	3.998-06	7.336-07	4.076-07	1.818-07	9.756-08
B	2.943-05	1.235-05	4.349-06	1.309-06	2.408-07	1.296-07
C	4.341-05	2.072-05	8.772-06	3.302-06	7.449-07	2.206-07
D	6.871-05	3.607-05	1.782-05	8.374-06	2.837-06	1.173-06
E	9.656-05	5.333-05	2.758-05	1.384-05	5.301-06	2.457-06
F	1.416-04	8.065-05	4.337-05	2.305-05	9.509-06	4.741-06
G	2.087-04	1.234-04	6.963-05	3.862-05	1.667-05	8.621-06

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	9.964-06	3.765-06	7.305-07	4.061-07	1.812-07	9.723-08
B	9.048-06	8.142-06	3.869-06	1.270-06	2.399-07	1.291-07
C	8.352-06	8.511-06	6.198-06	2.920-06	7.250-07	2.191-07
D	8.264-06	7.756-06	6.824-06	4.921-06	2.243-06	1.035-06
E	8.413-06	8.036-06	7.157-06	5.701-06	3.234-06	1.783-06
F	8.493-06	8.276-06	7.585-06	6.224-06	3.918-06	2.425-06
G	8.547-06	8.451-06	8.021-06	6.974-06	4.706-06	3.026-06

TABLE 9.2-2

FINITE CLOUD GAMMA TISSUE DOSE KERNEL

MEANDERING PLUME MODEL

Kr-83m

Kernel Units Are (mrad/hr)(m/sec)/(µCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	8.185-08	1.276-08	1.966-09	1.027-09	4.328-10	2.244-10
B	1.834-07	5.027-08	1.242-08	3.030-09	4.931-10	2.520-10
C	2.762-07	9.138-08	2.630-08	7.260-09	1.290-09	3.462-10
D	3.955-07	1.503-07	5.366-08	1.857-08	4.489-09	1.534-09
E	5.507-07	1.779-07	6.940-08	2.616-08	7.186-09	2.731-09
F	8.762-07	2.617-07	8.624-08	3.580-08	1.098-08	4.496-09
G	1.399-06	4.255-07	1.298-07	4.521-08	1.526-08	6.588-09

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	4.210-08	1.214-08	1.958-09	1.022-09	4.310-10	2.234-10
B	1.057-08	2.663-08	1.083-08	2.941-09	4.911-10	2.508-10
C	1.360-10	1.368-08	1.546-08	6.249-09	1.254-09	3.437-10
D	5.780-11	8.0[illegible]-11	4.648-09	6.766-09	3.193-09	1.313-09
E	5.228-11	3.9[illegible]6-11	4.249-10	2.623-09	2.802-09	1.627-09
F	4.922-11	3.163-11	2.106-11	1.306-10	8.106-10	9.258-10
G	4.820-11	2.911-11	1.586-11	9.434-12	6.733-12	8.527-11

TABLE 9.2-2 (Cont'd)

Kr-85m

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	3.663-06	7.804-07	1.275-07	6.744-08	2.858-08	1.483-08
B	6.256-06	2.450-06	7.454-07	1.957-07	3.256-08	1.665-03
C	8.066-06	3.729-06	1.386-[illegible]	4.412-07	8.389-08	2.283-08
D	9.878-06	5.066-06	2.234-06	[illegible]092-07	2.554-07	9.347-08
E	1.130-05	5.559-06	2.560-06	[illegible].113-06	3.562-07	1.477-07
F	1.354-05	6.467-06	2.822-06	[illegible]01-06	4.566-07	2.044-07
G	1.577-05	7.541-06	3.287-06	1.438-06	5.354-07	2.502-07

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	2.241-06	7.394-07	1.269-07	6.713-08	2.844-08	[illegible]6-08
B	1.899-06	1.576-06	6.603-07	1.901-07	3.240-08	[illegible].657-08
C	1.672-06	1.489-06	9.516-07	3.874-07	8.161-08	2.265-08
D	1.583-06	1.204-06	8.244-07	5.027-07	1.968-0[illegible]	8.175-08
E	1.584-06	1.163-06	7.158-07	4.283-07	2.026-[illegible]7	1.026-07
F	1.584-06	1.140-[illegible]6	6.583-07	3.563-07	1.661-[illegible]7	9.331-08
G	1.588-06	1.132-06	6.350-07	3.269-07	1.383-[illegible]7	7.433-08

TABLE 9.2-2 (Cont'd)

Kr-85

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	4.309-08	1.019-08	1.749-09	9.313-10	3.964-10	2.058-10
B	7.144-08	2.960-08	9.690-09	2.666-09	4.516-10	2.311-10
C	9.119-08	4.382-08	1.716-08	5.802-09	1.150-09	3.163-10
D	1.110-07	5.854-08	2.670-08	1.118-08	3.270-09	1.233-09
E	1.271-07	6.403-08	3.037-08	1.340-08	4.399-09	1.862-09
F	1.525-07	7.418-08	3.331-08	1.539-08	5.481-09	2.484-09
G	1.781-07	8.658-08	3.846-08	1.684-08	6.314-09	2.972-09

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	2.746-08	9.646-09	1.742-09	9.273-10	3.945-10	2.048-10
B	2.432-08	1.989-08	8.657-09	2.591-09	4.494-10	2.300-10
C	2.214-08	1.942-08	1.228-08	5.142-09	1.120-09	3.139-10
D	2.136-08	1.670-08	1.138-08	6.712-09	2.590-09	1.090-09
E	2.145-08	1.644-08	1.038-08	6.012-09	2.705-09	1.352-09
F	2.152-08	1.631-08	9.866-09	5.302-09	2.354-09	1.278-09
G	2.160-08	1.628-08	9.665-09	5.022-09	2.079-09	1.090-09

TABLE 9.2-2 (Cont'd)

Kr-87

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.215-05	3.144-06	5.931-07	3.189-07	1.377-07	7.171-08
B	1.989-05	8.546-06	2.983-06	8.821-07	1.569-07	8.051-08
C	2.530-05	1.242-05	5.094-06	1.811-06	3.868-07	1.096-07
D	3.074-05	1.644-05	7.721-06	3.317-06	1.002-06	3.893-07
E	3.525-05	1.795-05	8.737-06	3.924-06	1.305-06	5.627-07
F	4.236-05	2.078-05	9.554-06	4.469-06	1.588-06	7.274-07
G	4.960-05	2.426-05	1.099-05	4.867-06	1.800-06	8.530-07

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	7.911-06	2.984-06	5.904-07	3.175-07	1.371-07	7.136-08
B	7.082-06	5.897-06	2.688-06	8.585-07	1.561-07	8.012-08
C	6.497-06	5.794-06	3.754-06	1.625-06	3.772-07	1.088-07
D	6.285-06	5.062-06	3.548-06	2.110-06	8.163-07	3.488-07
E	6.314-06	5.008-06	3.295-06	1.939-06	8.554-07	4.252-07
F	6.337-06	4.982-06	3.168-06	1.757-06	7.692-07	4.038-07
G	6.362-06	4.980-06	3.121-06	1.616-06	6.987-07	3.625-07

TABLE 9.2-2 (Cont'd)

Kr-88

Kernel Units Are (mrad/hr)(m/sec)(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	2.753-05	7.440-06	1.449-06	7.821-07	3.391-07	1.766-07
B	4.481-05	1.961-05	7.044-06	2.141-06	3.861-07	1.983-07
C	5.690-05	2.824-05	1.183-05	4.300-06	9.437-07	2.696-07
D	6.912-05	3.720-05	1.772-05	7.695-06	2.361-06	9.282-07
E	7.940-05	4.060-05	2.001-05	9.049-06	3.032-06	1.318-06
F	9.571-05	4.700-05	2.185-05	1.026-05	3.647-06	1.679-06
G	1.125-04	5.500-05	2.509-05	1.115-05	4.107-06	1.952-06

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.814-05	7.066-06	1.442-06	7.787-07	3.375-07	1.758-07
B	1.633-05	1.371-05	6.374-06	2.085-06	3.843-07	1.974-07
C	1.504-05	1.351-05	8.841-06	3.876-06	9.209-07	2.675-07
D	1.458-05	1.191-05	8.441-06	5.027-06	1.945-06	8.359-07
E	1.465-05	1.181-05	7.904-06	4.673-06	2.042-06	1.013-06
F	1.471-05	1.177-05	7.644-06	4.282-06	1.861-06	9.839-07
G	1.477-05	1.177-05	7.551-06	4.132-06	1.703-06	8.830-07

TABLE 9.2-2 (Cont'd)

Kr-89

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	2.621-05	6.844-06	1.296-06	6.973-07	3.013-07	1.569-07
B	4.285-05	1.849-05	6.486-06	1.926-06	3.432-07	1.761-07
C	5.446-05	2.682-05	1.104-05	3.945-06	8.451-07	2.397-07
D	6.614-05	3.545-05	1.670-05	7.187-06	2.178-06	8.479-07
E	7.587-05	3.871-05	1.889-05	8.488-06	2.830-06	1.222-06
F	9.121-05	4.480-05	2.065-05	9.655-06	3.434-06	1.575-06
G	1.068-04	5.233-05	2.374-05	1.050-05	3.889-06	1.844-06

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.712-05	6.495-06	1.290-06	6.943-07	2.999-07	1.561-07
B	1.535-05	1.280-05	5.850-06	1.875-06	3.415-07	1.753-07
C	1.411-05	1.259-05	8.162-06	3.541-06	8.243-07	2.379-07
D	1.366-05	1.103-05	7.738-06	4.597-06	1.778-06	7.603-07
E	1.372-05	1.092-05	7.202-06	4.237-06	1.864-06	9.261-07
F	1.377-05	1.087-05	6.936-06	3.849-06	1.681-06	8.942-07
G	1.383-05	1.087-05	6.839-06	3.698-06	1.531-06	7.937-07

TABLE 9.2-2 (Cont'd)

Kr-90

Kernel Units Are (mrad/hr)(m/sec)/(µCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	3.596-05	9.220-06	1.717-06	9.224-07	3.975-07	2.068-07
B	5.904-05	2.524-05	8.737-06	2.564-06	4.526-07	2.322-07
C	7.515-05	3.677-05	1.499-05	5.297-06	1.122-06	3.164-07
D	9.136-05	4.874-05	2.281-05	9.756-06	2.337-06	1.137-06
E	1.048-04	5.325-05	2.584-05	1.156-05	3.842-06	1.652-06
F	1.260-04	6.166-05	2.827-05	1.318-05	4.688-06	2.145-06
G	1.475-04	7.205-05	3.254-05	1.436-05	5.329-06	2.522-06

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	2.336-05	8.747-06	1.709-06	9.183-07	3.956-07	2.058-07
B	2.084-05	1.736-05	7.868-06	2.495-06	4.505-07	2.311-07
C	1.908-05	1.702-05	1.101-05	4.746-06	1.094-06	3.139-07
D	1.845-05	1.483-05	1.038-05	6.165-06	2.385-06	1.017-06
E	1.853-05	1.465-05	9.611-06	5.650-06	2.498-06	1.243-06
F	1.859-05	1.457-05	9.230-06	5.102-06	2.236-06	1.194-06
G	1.867-05	1.456-05	9.088-06	4.889-06	2.024-06	1.051-06

TABLE 9.2-2 (Cont'd)

Xe-131m

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	5.987-07	1.028-07	1.607-08	8.426-09	3.558-09	1.845-09
B	1.193-06	3.769-07	9.946-08	2.476-08	4.054-09	2.072-09
C	1.662-06	6.382-07	2.019-07	5.831-08	1.056-08	2.845-09
D	2.200-06	9.527-07	3.740-07	1.383-07	3.529-08	1.232-08
E	2.702-06	1.079-06	4.544-07	1.836-07	5.378-08	2.111-08
F	3.547-06	1.377-06	5.291-07	2.323-07	7.627-08	3.249-08
G	4.454-06	1.786-06	6.851-07	2.730-07	9.724-08	4.374-08

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	3.259-07	9.764-08	1.601-08	8.388-09	3.541-09	1.836-09
B	1.737-07	2.145-07	8.709-08	2.402-08	4.034-09	2.061-09
C	9.738-08	1.544-07	1.255-07	5.047-08	1.027-08	2.822-09
D	7.551-08	6.365-08	6.997-08	5.984-08	2.577-08	1.062-08
E	7.247-08	5.216-08	3.944-08	3.702-08	2.443-08	1.328-08
F	7.054-08	4.733-08	2.830-08	1.861-08	1.337-08	9.664-09
G	6.993-08	4.547-08	2.491-08	1.353-08	6.619-09	4.415-09

TABLE 9.2-2 (Cont'd)

Xe-133m

Kernel Units Are (mrad/hr)(m/sec)(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.081-06	2.077-07	3.334-08	1.757-08	7.435-09	3.857-09
B	2.015-06	7.008-07	1.996-07	5.123-08	8.470-09	4.331-09
C	2.722-06	1.130-06	3.864-07	1.176-07	2.192-08	5.942-09
D	3.500-06	1.620-06	6.690-07	2.591-07	6.947-08	2.490-08
E	4.194-06	1.811-06	7.925-07	3.311-07	1.011-07	4.081-08
F	5.342-06	2.230-06	9.021-07	4.043-07	1.367-07	5.958-08
G	6.550-06	2.785-06	1.120-06	4.631-07	1.680-07	7.682-08

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	6.234-07	1.970-07	3.320-08	1.749-08	7.399-09	3.838-09
B	4.269-07	4.246-07	1.758-07	4.974-08	8.429-09	4.309-09
C	3.214-07	3.541-07	2.530-07	1.027-07	2.132-08	5.895-09
D	2.886-07	2.268-07	1.817-07	1.276-07	5.224-08	2.164-08
E	2.859-07	2.106-07	1.375-07	9.525-08	5.175-08	2.708-08
F	2.842-07	2.032-07	1.192-07	6.762-08	3.599-08	2.229-08
G	2.841-07	2.005-07	1.130-07	5.889-08	2.568-08	1.461-08

TABLE 9.2-2 (Cont'd)

Xe-133

Kernel Units Are (mrad/hr)(m/sec)/(µCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.244-06	2.319-07	3.668-08	1.929-08	8.152-09	4.228-09
B	2.309-06	8.023-07	2.231-07	5.647-08	9.287-09	4.748-09
C	3.102-06	1.294-06	4.386-07	1.312-07	2.412-08	6.516-09
D	3.952-06	1.847-06	7.633-07	2.943-07	7.809-08	2.773-08
E	4.675-06	2.059-06	9.015-07	3.765-07	1.148-07	4.611-08
F	5.842-06	2.504-06	1.021-06	4.591-07	1.555-07	6.777-08
G	7.019-06	3.065-06	1.254-06	5.238-07	1.906-07	8.731-08

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	7.099-07	2.200-07	3.654-08	1.920-08	8.113-09	4.208-09
B	4.924-07	4.808-07	1.962-07	5.481-08	9.242-09	4.725-09
C	3.697-07	4.007-07	2.838-07	1.141-07	2.345-08	6.465-09
D	3.266-07	2.521-07	2.012-07	1.419-07	5.819-08	2.401-08
E	3.221-07	2.294-07	1.483-07	1.047-07	5.755-08	3.020-08
F	3.190-07	2.183-07	1.245-07	7.202-08	3.945-08	2.466-08
G	3.185-07	2.139-07	1.158-07	6.066-08	2.711-08	1.573-08

TABLE 9.2-2 (Cont'd)

Xe-135m

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	8.424-06	1.988-06	3.415-07	1.818-07	7.740-08	4.019-08
B	1.401-05	5.783-06	1.890-06	5.204-07	8.818-08	4.512-08
C	1.792-05	8.575-06	3.351-06	1.132-06	2.244-07	6.177-08
D	2.185-05	1.148-05	5.225-06	2.185-06	6.382-07	2.407-07
E	2.507-05	1.256-05	5.949-06	2.621-06	8.595-07	3.636-07
F	3.016-05	1.459-05	6.532-06	3.015-06	1.072-06	4.857-07
G	3.530-05	1.707-05	7.560-06	3.302-06	1.237-06	5.819-07

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	5.360-06	1.882-06	3.400-07	1.810-07	7.703-08	3.999-08
B	4.720-06	3.879-06	1.689-06	5.058-07	8.775-08	4.490-08
C	4.284-06	3.776-06	2.396-06	1.003-06	2.186-07	6.128-08
D	4.129-06	3.232-06	2.210-06	1.308-06	5.053-07	2.127-07
E	4.146-06	3.181-06	2.012-06	1.169-06	5.27[illegible]-07	2.637-07
F	4.158-06	3.155-06	1.911-06	1.028-06	4.574-07	2.487-07
G	4.174-06	3.148-06	1.872-06	9.735-07	4.032-[illegible]	2.116-07

TABLE 9.2-2 (Cont'd)

Xe-135

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	5.423-06	1.197-06	1.983-07	1.051-07	4.458-08	2.313-08
B	9.150-06	3.656-06	1.142-06	3.038-07	5.079-08	2.598-08
C	1.175-05	5.514-06	2.090-06	6.778-07	1.305-07	3.560-08
D	1.435-05	7.439-06	3.316-06	1.366-06	3.89[illegible]-07	1.437-07
E	1.640-05	8.151-06	3.789-06	1.659-06	5.357-07	2.237-07
F	1.962-05	9.457-06	4.168-06	1.926-06	6.796-07	3.056-07
G	2.281-05	1.100-05	4.835-06	2.121-06	7.917-07	3.709-07

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	3.368-06	1.134-06	1.974-07	1.046-07	4.437-08	2.302-08
B	2.913-06	2.389-06	1.014-06	2.951-07	5.054-08	2.585-08
C	2.603-06	2.294-06	1.455-06	5.969-07	1.269-07	3.533-08
D	2.485-06	1.906-06	1.295-06	7.775-07	3.024-07	1.261-07
E	2.491-06	1.856-06	1.148-06	6.758-07	3.133-07	1.578-07
F	2.495-06	1.829-06	1.071-06	5.759-07	2.631-07	1.457-07
G	2.502-06	1.820-06	1.040-06	5.355-07	2.244-07	1.194-07

TABLE 9.2-2 (Cont'd)

Xe-137

Kernel Units Are (mrad/hr)(m/sec)(µCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	3.689-06	8.835-07	1.544-07	8.235-08	3.516-08	1.826-08
B	6.111-06	2.542-06	8.399-07	2.342-07	4.005-08	2.051-08
C	7.800-06	3.756-06	1.481-06	5.040-07	1.013-07	2.804-08
D	9.494-06	5.013-06	2.295-06	9.647-07	2.834-07	1.074-07
E	1.088-05	5.482-06	2.609-06	1.154-06	3.795-07	1.610-07
F	1.305-05	6.351-06	2.860-06	1.325-06	4.712-07	2.139-07
G	1.524-05	7.411-06	3.302-06	1.449-06	5.417-07	2.552-07

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	2.356-06	8.370-07	1.537-07	8.199-08	3.499-08	1.818-08
B	2.086-06	1.713-06	7.513-07	2.276-07	3.985-08	2.041-08
C	1.899-06	1.673-06	1.064-06	4.476-07	9.868-08	2.782-08
D	1.831-06	1.438-06	9.862-07	5.835-07	2.254-07	9.512-08
E	1.839-06	1.417-06	9.011-07	5.244-07	2.355-07	1.176-07
F	1.845-06	1.405-06	8.573-07	4.641-07	2.058-07	1.114-07
G	1.852-06	1.403-06	8.403-07	4.402-07	1.824-07	9.554-08

TABLE 9.2-2 (Cont'd)

Xe-138

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.708-05	4.434-06	8.332-07	4.483-07	1.933-07	1.006-07
B	2.800-05	1.203-05	4.200-06	1.243-06	2.201-07	1.129-07
C	3.563-05	1.748-05	7.175-06	2.547-06	5.444-07	1.538-07
D	4.334-05	2.314-05	1.087-05	4.664-06	1.411-06	5.479-07
E	4.972-05	2.528-05	1.230-05	5.520-06	1.838-06	7.922-07
F	5.984-05	2.927-05	1.345-05	6.289-06	2.235-06	1.024-06
G	7.018-05	3.420-05	1.548-05	6.851-06	2.536-06	1.201-06

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.113-05	4.208-06	8.294-07	4.463-07	1.924-07	1.001-07
B	9.951-06	8.303-06	3.786-06	1.209-06	2.191-07	1.124-07
C	9.124-06	8.151-06	5.289-06	2.286-06	5.309-07	1.527-07
D	8.823-06	7.117-06	4.996-06	2.970-06	1.149-06	4.909-07
E	8.864-06	7.040-06	4.640-06	2.732-06	1.204-06	5.986-07
F	8.895-06	7.004-06	4.462-06	2.475-06	1.082-06	5.767-07
G	8.929-06	7.000-06	4.397-06	2.376-06	9.825-07	5.099-07

TABLE 9.2-2 (Cont'd)

Ar-41

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	2.008-05	5.251-06	9.794-07	5.266-07	2.267-07	1.179-07
B	3.278-05	1.417-05	4.961-06	1.465-06	2.581-07	1.324-07
C	4.166-05	2.053-05	8.453-06	3.011-06	6.406-07	1.805-07
D	5.061-05	2.713-05	1.278-05	5.488-06	1.664-06	6.468-07
E	5.810-05	2.963-05	1.446-05	6.476-06	2.165-06	9.340-07
F	6.993-05	3.430-05	1.581-05	7.361-06	2.629-06	1.205-06
G	8.201-05	4.012-05	1.818-05	8.005-06	2.980-06	1.412-06

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.314-05	4.982-06	9.750-07	5.243-07	2.256-07	1.173-07
B	1.180-05	9.820-06	4.473-06	1.426-06	2.569-07	1.317-07
C	1.085-05	9.673-06	6.253-06	2.702-06	6.247-07	1.791-07
D	1.052-05	8.496-06	5.947-06	3.515-06	1.357-06	5.793-07
E	1.057-05	8.420-06	5.544-06	3.243-06	1.424-06	7.071-07
F	1.061-05	8.386-06	5.346-06	2.950-06	1.284-06	6.829-07
G	1.066-05	8.386-06	5.274-06	2.837-06	1.169-06	6.059-07

TABLE 9.2-2 (Cont'd)

0.8 MeV

Kernel Units Are (mrad/hr)(m/sec)/(μCi/sec)

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	1.418-05	3.529-06	6.249-07	3.340-07	1.427-07	7.413-08
B	2.331-05	9.872-06	3.333-06	9.467-07	1.625-07	8.323-08
C	2.967-05	1.445-05	5.797-06	2.010-06	4.101-07	1.138-07
D	3.604-05	1.920-05	8.901-06	3.765-06	1.121-06	4.288-07
E	4.135-05	2.098-05	1.010-05	4.470-06	1.485-06	6.342-07
F	4.967-05	2.431-05	1.106-05	5.101-06	1.828-06	8.328-07
G	5.806-05	2.842-05	1.275-05	5.557-06	2.091-06	9.869-07

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	9.169-06	3.344-06	6.222-07	3.325-07	1.420-07	7.377-08
B	8.182-06	6.742-06	2.990-06	9.206-07	1.617-07	8.283-08
C	7.494-06	6.620-06	4.221-06	1.792-06	3.996-07	1.129-07
D	7.253-06	5.764-06	3.974-06	2.336-06	9.005-07	3.812-07
E	7.287-06	5.700-06	3.671-06	2.126-06	9.436-07	4.698-07
F	7.314-06	5.669-06	3.519-06	1.908-06	8.360-07	4.491-07
G	7.344-06	5.664-06	3.462-06	1.823-06	7.509-07	3.911-07

TABLE 9.2-3

POPULATION DISTRIBUTION

DRESDEN STATION*

SECTOR	RANGE, METERS				
	400-800	800-1,609	1,609-3,218	3,218-8,045	8,045-16,090
N	0	0	3	789	303
NNE	0	0	3	186	2,763
NE	0	0	9	1,979	1,929
ENE	0	6	6	0	743
E	0	12	0	12	1,196
ESE	0	15	33	15	573
SE	0	15	279	810	4,655
SSE	0	96	6	63	320
S	0	0	6	18	3,750
SSW	0	0	6	207	1,313
SW	0	0	0	48	288
WSW	0	0	0	27	68
W	0	0	3	33	8,118
WNW	0	0	0	48	657
NW	0	0	6	303	407
NNW	0	3	6	39	301

* Environmental Report, Supplement 1, Appendix A.

TABLE 9.2-3 (Cont'd)

LASALLE COUNTY STATION*

SECTOR	RANGE, METERS				
	400-800	800-1,609	1,609-3,218	3,218-8,045	8,045-16,090
N	0	0	3	7	2,014
NNE	0	0	0	63	1,078
NE	0	0	0	370	1,119
ENE	0	0	0	25	260
E	0	0	3	44	560
ESE	0	0	0	41	350
SE	0	0	3	36	275
SSE	0	0	0	49	526
S	0	0	0	39	512
SSW	0	3	9	49	475
SW	0	0	3	25	1,299
WSW	0	3	18	31	269
W	0	0	15	39	753
WNW	0	0	13	32	965
NW	0	0	0	113	1,703
NNW	0	0	6	48	3,339

* Environmental Report, OL Stage, Figures 2.1-6 and 2.1-7.

TABLE 9.2-3 (Cont'd)

QUAD-CITIES STATION*

SECTOR	RANGE, METERS				
	400-800	800-1,609	1,609-3,218	3,218-8,045	8,045-16,090
N	0	14	45	315	641
NNE	0	2	25	2,050	7,799
NE	0	2	15	315	217
ENE	0	7	5	215	582
E	0	0	0	22	412
ESE	0	0	0	23	610
SE	0	10	10	95	420
SSE	0	25	25	570	629
S	0	0	10	350	2,867
SSW	0	0	5	845	3,170
SW	0	[illegible]	0	195	425
WSW	0	0	0	120	315
W	0	0	0	240	610
WNW	0	0	0	165	847
NW	0	0	0	65	630
NNW	0	0	0	65	710

* Figure 3, Environmental Report, Supplement 3; 1976 data.

TABLE 9.2-3 (Cont'd)

ZION STATION*

SECTOR	RANGE, METERS				
	400-800	800-1,609	1,609-3,218	3,218-8,045	8,045-16,090
N	0	99	353	1,954	19,934
NNE	0	0	0	0	0
NE	0	0	0	0	0
ENE	0	0	0	0	0
E	0	0	0	0	0
ESE	0	0	0	0	0
SE	0	0	0	0	0
SSE	0	0	0	0	0
S	0	0	0	1,320	70,146
SSW	0	0	999	10,995	77,242
SW	0	0	1,264	12,514	18,768
WSW	0	204	847	9,956	2,643
W	0	440	1,349	10,069	874
WNW	0	302	2,571	6,310	521
NW	0	64	1,635	7,918	1,415
NNW	0	82	452	4,841	33,749

* 1975 Estimate Based on Question Q2.4, Amendment 14 to the FSAR, August 14, 1971.

TABLE 9.2-4

RELATIVE EFFLUENT CONCENTRATION, STRAIGHT-LINE GAUSSIAN PLUME MODEL

$(u\chi/Q)$, $1/m^2$

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	4.970-05	7.794-06	1.104-06	5.910-07	2.585-07	1.382-07
B	1.062-04	3.068-05	6.305-06	8.970-07	3.437-07	1.838-07
C	1.990-04	6.735-05	1.984-05	6.010-06	1.255-06	4.066-07
D	3.607-04	1.552-04	5.739-05	2.075-05	5.406-06	1.969-06
E	5.164-04	2.632-04	1.084-04	4.241-05	1.241-05	4.994-06
F	9.523-04	4.246-04	2.018-04	8.758-05	2.835-05	1.219-05
G	2.625-03	8.238-04	3.879-04	1.945-04	6.974-05	3.127-05

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	2.116-05	7.322-06	1.101-06	5.885-07	2.572-07	1.376-07
B	8.207-06	1.729-05	5.781-06	8.922-07	3.421-07	1.829-07
C	2.097-07	9.773-06	1.396-05	5.103-06	1.210-06	4.011-07
D	4.313-13	1.621-07	4.670-06	7.462-06	3.809-06	1.678-06
E	1.082-21	1.353-10	4.944-07	3.904-06	5.149-06	2.925-06
F	0.000	1.676-19	6.203-10	2.357-07	1.933-06	2.410-06
G	0.000	0.000	1.150-19	1.193-11	3.820-08	3.342-07

TABLE 9.2-5

RELATIVE EFFLUENT CONCENTRATION, MEANDERING PLUME MODEL

(uX/Q), 1/m²

RELEASE HEIGHT = 0. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	6.647-05	9.587-06	1.263-06	6.313-07	2.525-07	1.263-07
B	1.087-04	2.853-05	5.433-06	7.209-07	2.525-07	1.263-07
C	1.589-04	4.796-05	1.302-05	3.671-06	7.006-07	2.121-07
D	2.156-04	7.940-05	2.666-05	8.941-06	2.125-06	7.236-07
E	2.675-04	9.932-05	3.620-05	1.304-05	3.472-06	1.305-06
F	4.254-04	1.248-04	4.768-05	1.871-05	5.481-06	2.199-06
G	7.033-04	2.064-04	6.228-05	2.552-05	8.126-06	3.389-06

RELEASE HEIGHT = 100. (METERS)

STABILITY CLASS	DOWNWIND DISTANCE(METERS) 400.	800.	1609.	3218.	8045.	16090.
A	2.756-05	8.946-06	1.257 06	6.283-07	2.513-07	1.257-07
B	8.065-06	1.609-05	4.964-06	[illegible].164-07	2.513-07	1.257-07
C	1.565-07	6.819-06	9.251-06	3.111-06	6.749-07	2.092-07
D	2.266-13	7.967-08	2.144-06	3.236-06	1.496-06	6.162-07
E	4.042-22	4.726-11	1.614-07	1.192-06	1.471-06	7.640-07
F	0.000	4.042-20	1.398-10	4.966-08	3.727-07	4.402-07
G	0.000	0.000	1.555-20	1.507-12	4.417-09	2.723-08

TABLE 9.2-6

SPLIT SIGMA CORRECTION FACTORS, $C(\sigma_\theta,\Delta T)$

STABILITY CLASS DETERMINED BY DIFFERENTIAL TEMPERATURE, ΔT*	STABILITY CLASS DETERMINED BY HORIZONTAL VARIATION, σ_θ*						
	A	B	C	D	E	F	G
A	1.0	1.330	1.751	2.487	3.497	5.067	7.605
B	0.752	1.0	1.317	1.870	2.630	3.810	5.719
C	0.571	0.759	1.0	1.420	1.997	2.893	4.343
D	0.402	0.535	0.704	1.0	1.406	2.037	3.058
E	0.286	0.380	0.501	0.711	1.0	1.449	2.175
F	0.197	0.263	0.346	0.491	0.690	1.0	1.501
G	0.132	0.175	0.230	0.327	0.460	0.666	1.0

* See Table 7.1-5.

9.3 RADIOIODINES, "PARTICULATES", AND OTHER (NONNOBLE GAS) RADIONUCLIDES

9.3.1 Inhalation Dose

The method of calculating the dose rate to internal organs due to inhalation of airborne radioactive material is described in Subsection 2.1.2 of this manual. Presently, only ground level or elevated releases will be considered. Following Equation 2.18, then, the present case may be written as follows:

$$D'_{ja} = \frac{10^6}{8760} R_a \sum_i DFA_{ija} \left[\left({}^{X}/_{Q} \right)_s Q'_{is} + \left({}^{X}/_{Q} \right)_g Q'_{ig} \right] \tag{9.27}$$

D'_{ja}	Dose Rate The dose rate to organ j, age group a.	(mrem/hr)
10^6	Conversion Constant Converts µCi to pCi.	(µCi/pCi)
R_a	Inhalation Rate The inhalation rate for individuals of age group a.	(m^3/yr)
DFA_{ija}	Inhalation Dose Factor The inhalation dose commitment factor for radionuclide i, organ j, and age group a. See Table 7.1-1.	(mrem/pCi)

Combining Equation 9.27 with Equations 9.10, 9.11, 9.20, and 9.21 (and 9.22, if applicable) results in the final equation as follows:

$$D'_{ja} = \frac{10^6}{8760} R_a \sum_i DFA_{ija} \left[\frac{1}{u_s} \left({}^{uX}/_{Q} \right)_s Q_{is} \exp(-\lambda_i R/3600u_s) + \frac{1}{u_g} \left({}^{uX}/_{Q} \right)_g Q_{ig} \exp(-\lambda_i R/3600u_g) \right] \tag{9.28}$$

If one has an estimate of the release rate Q, then, using Equation 9.28, the information from the meteorological tower, and the data of Tables 7.1-1 and 9.2-4 or 9.2-5 (as appropriate), one can quickly estimate downwind inhalation dose rates.

(NOTE...The "dose rate" determined by Equation 9.28 is not an instantaneous dose rate but the dose commitment (mrem) received per hour of exposure. The actual dose delivered to the organ of interest is delivered over a period of time which is dependent on metabolism, radiodecay half-life, and biological half-life.)

An offsite dose (dose commitment) may be estimated from the following:

$$D_{ja} = \frac{10^6}{8760 \times 3600} R_a \sum_i DFA_{ija} \left[\frac{1}{u_s} \left(u\chi/Q \right)_s A_{is} \exp(-\lambda_i R/3600 u_s) + \frac{1}{u_g} \left(u\chi/Q \right)_g A_{ig} \exp(-\lambda_i R/3600 u_g) \right] \quad (9.29)$$

D_{ja}	Inhalation Dose The time integrated dose to organ j, age group a, caused by inhalation of airborne radionuclides.	(mrem)
A_{is}	Accumulative Release, Stack The accumulative release from the stack of nuclide i over the time period of interest.	(μCi)
A_{ig}	Accumulative Release, Ground Level The accumulative release, at ground level, of nuclide i over the time period of interest.	(μCi)

9.3.2 Symbols Used in Section 9.3

SYMBOL	NAME	UNIT
D'_{ja}	Dose Rate	(mrem/hr)
R_a	Inhalation Rate	(m^3/yr)
DFA_{ija}	Inhalation Dose Factor	(mrem/pCi)
$(\chi/Q)_s$	Relative Effluent Concentration, Stack Release	(sec/m^3)
$(\chi/Q)_g$	Relative Effluent Concentration, Ground Level	(sec/m^3)
Q'_{is}	Release Rate From Stack, Adjusted For Radiodecay	(μCi/sec)
Q'_{ig}	Release Rate at Ground Level, Adjusted for Radiodecay	(μCi/sec)
u	Wind Speed	(m/sec)
u_s	Wind Speed, Stack Elevation	(m/sec)
$(u\chi/Q)_s$	Dispersion Factor, Stack Release	(1/m^2)
Q_{is}	Release Rate from Stack	(μCi/sec)
R	Downwind Distance	(m)
u_g	Wind Speed, Ground Level	(m/sec)
$(u\chi/Q)_g$	Dispersion Factor, Ground Release	(1/m^2)
Q_{ig}	Release Rate at Ground Level	(μCi/sec)
λ_i	Radiodecay Constant	(1/hr)
D_{ja}	Inhalation Dose	(mrem)
A_{is}	Accumulative Release, Stack	(μCi)
A_{ig}	Accumulative Release, Ground Level	(μCi)

9.3.3 Constants Used in Section 9.3

NUMERICAL VALUE	NAME	UNIT
10^6	Conversion Constant	(μCi/pCi)
8760	Conversion Constant	(hr/yr)
3600	Conversion Constant	(sec/hr)

9.4 LAKE BREEZE EFFECTS (ZION STATION ONLY)

Currently recommended meteorological programs and diffusion methods for nuclear power plants located in coastal zones were recently reviewed for the U.S. Nuclear Regulatory Commission (Reference 6.20, NUREG-0936). Among certain deficiencies in guidelines and procedures noted in this document were "failure to consider the role of coastal internal boundary layers, specifications for tower locations and instrument heights, (and) methods for classifying atmospheric stability...." Included were recommendations for changes to the guidelines.

An atmospheric dispersion model has been developed to account for boundary layer conditions that could occur at the Zion plant. The model development essentially followed the various methods itemized in the reference cited. Conservatively high ground level concentrations result from the model when compared to standard dispersion calculations.

9.4.1 The Boundary Layer

Continuous measurements of the boundary layer in the vicinity of the Zion Plant are not available. Indeed, aside from a few intensive short-term studies of lake shore dispersion in the vicinity, no boundary layer data exist. Consequently, readily available meteorological measurements representing a 2-year period were used in conjunction with the boundary Equation (1), found in NUREG-0936, to infer the existence and location of the boundary.

The equation was evaluated subject to the following assumptions and conditions:

a. friction velocity $u^* = 1$ mps;
b. wind speed of at least 6 mps;
c. land-water temperature contrast at least 5° F;

d. air mass stability was estimated by the 250- to 125-foot differential temperature measured on the Zion tower; and
e. wind direction was onshore.

The results are shown in Figure 9.4-1. In summary, the boundary was computed to occur roughly 10% of the hours annually (876/8760). Of those hours it occurred well above the release point 95% of the time (832 hours). The remaining 10% of the time (44 hours) it was below the release point leading to potential fumigation downwind (cf. Table 9.4-1).

It should be noted that the existing meteorological tower is located entirely within the calculated boundary. For all practical cases, then, the measurements from the tower can be assumed to represent the boundary layer conditions and not be partway in the boundary layer and partway in the "lake" air (a caution referred to in NUREG-0936).

9.4.2 Dispersion Model

9.4.2.1 Dispersion Conditions

When a boundary height, variable both in time and inland fetch, is taken into account, four downwind zones with different dispersion characteristics emerge. The dispersion equations differ for the four cases summarized below (cf. Figure 9.4-2):

Case 1. The boundary layer is located above the release point. Consequently, vertical dispersion is limited by the boundary and the ground at all ranges downwind to 10 miles (the downwind extent of the model evaluation). Boundary layer dispersion is characterized by meteorological tower measurements (Figure 9.4-2, Case 1).

Case 2. The boundary layer is located below the release point. This can lead to three distinct cases depending on the downwind range in question (Figure 9.4-2, Case 2).

Case 2.1 Dispersion at downwind distances from the release point to the point X_1, beneath which the bottom of the plume intersects the boundary. The plume is embedded in the relatively turbulent-free lake air.

Case 2.2 Dispersion at downwind distances from point X_1 to the point X_2, beneath which the top of the plume intersects the boundary. In this zone, fumigation is assumed to occur. The effluent is uniformly distributed in the vertical.

Case 2.3 Dispersion at downwind distances beyond the point X_2. Here, limited mixing occurs due to the plume being trapped beneath the boundary. Here also the effluent is uniformly distributed in the vertical.

9.4.2.2 Results

The model was evaluated at various downwind distances to 10 miles, to yield the "worst case" values. The highest concentrations were due either to Case 1 or Case 2.2. The remaining cases were, therefore, eliminated as possible worst cases.

9.4.2.3 Required Forecast Inputs to Model

The lake effects model requires a variety of inputs. Some are used to determine whether or not a boundary exists.

Others are used to select the limited mixing or the fumigation mode. The inputs used to decide whether lake effects will occur are:

a. hour of day,
b. wind direction,
c. wind speed, and
d. temperature contrast between lake and land.

The additional input used to select the appropriate dispersion mode is air mass stability.

9.4.2.3.1 Hour Of Day

The internal boundary is assumed to develop, in part, in response to heating being transferred upward from the earth's surface into the air. Thus, the time of day is limited to those daylight hours beginning several hours after sunrise to late afternoon. It is during this time that the sun has its greatest effect and air temperature near the ground reaches its maximum.

9.4.2.3.2 Wind Direction

The populated areas subject to an accidental release are restricted to the landward region exclusive of the lake. Consequently, the wind must flow onshore before an overland internal boundary can develop.

9.4.2.3.3 Wind Speed

The boundary forms only after an onshore wind has developed, and cannot form in the absence of wind. Thus, some minimum speed greater than zero is necessary for the formation to occur. Moreover, field studies have suggested that optimum conditions for a boundary to mature imply a maximum wind speed beyond which the boundary presumably cannot be maintained. Thus, the wind speed seems to be constrained to fall between both a lower and an upper bound.

9.4.2.3.4 Temperature Contrast Between Lake and Land

The temperature differential between the air over the water and that over land precedes the formation of the boundary and appears to be a major factor in the vertical slope of the boundary. Nominal temperature differentials in the Chicago-Zion area have been found to be on the order of 6° to 7° Celsius. It is probable that there is some value below which the boundary will not form. Since it is not known at this time, a value equal to approximately one-half of the nominal differential is used in the present model.

The additional model input used to select the appropriate dispersion mode is air mass stability. Field studies have shown that the optimum condition is near neutral (dry adiabatic lapse rate).

Boundary formations appear to be discouraged under extreme conditions of stability. It should be noted that the air mass parameter referred to here represents the synoptic scale and not the underlying boundary layer.

Daytime hours are those beginning 1 hour after sunrise and ending 1 hour before sunset (CST). Wind speed and direction are taken directly from the 35-foot level of the meteorological tower. These signals will be hardwired into the process computer. Signals representing the temperature differential between the lake and land and air mass stability are not directly available. Instead they are determined from a variety of meteorological reporting stations and provided by the meteorological consultant. Predicted hourly differentials and stability factors are also prepared by the consultant.

Precalculated "worst case" values of ground level concentrations have been tabulated and stored in the process computer for immediate access. The appropriate value is indicated by the meteorological data (actual or forecast). The results are listed in Table 9.4-2.

9.4.3 Dose Models

For simplicity of programming, the Zion lake breeze model uses a semi-infinite plume model in place of the finite plume model previously discussed. Both the whole body dose and skin dose calculations are described using this method.

9.4.3.1 Whole Body Dose from Noble Gases

A whole body dose may be calculated using the precalculated "worst case" values of relative ground level concentration, whole body dose factors, and actual plant emissions. The equation becomes:

$$D_\gamma(R) = 3.17 \times 10^{-8} \sum_i (\chi/Q)\ A_i\ \bar{X}_i \exp\left(-\lambda_i R/3600\ u_g\right) \quad (9.30)$$

$D_\gamma(R)$	Whole Body Gamma Ray Dose Whole body gamma ray dose at the downwind distance R.	(mrem)
3.17×10^{-8}	Conversion Constant Converts seconds to years.	(years/second)
(χ/Q)	Relative Effluent Concentration The relative effluent concentration at ground level.	(sec/m^3)
A_i	Accumulative Release The accumulative release of nuclide i over the time period of interest.	(µCi)

$\overline{X}_i$ Whole Body Gamma Dose Factor (mrem/yr per $\mu Ci/m^3$)

The whole body gamma dose factor for a semi-infinite cloud of the radionuclide i. Values are found in Table 7.1-13.

λ_i Radiodecay Constant (1/hr)

The radioactive decay constant for nuclide i. See Table 7.1-9.

u_g Wind Speed, Ground Level (m/sec)

The wind speed at the lowest position on the meteorological tower.

In cases where σ_θ values are used to infer horizontal plume width, given in Table 9.4-2, the χ/Q values must be multiplied by the appropriate factor given in Table 9.2-6. These factors are independent of the lake breeze model and should be applied any time σ_θ values are used.

9.4.3.2 Skin Dose from Noble Gases

A skin dose may be calculated using the precalculated "worst case" values of relative ground level concentrations, beta skin dose factors, gamma air dose factors, and actual plant emissions. The equation becomes:

$$D_{skin}(R) = 3.17 \times 10^{-8} \; (\chi/Q) \sum_i A_i (X_i + \overline{L}_i) \exp(-\lambda_i R/3600 \, u_g) \qquad (9.31)$$

$D_{skin}(R)$	Skin Dose at Distance R.	(mrem)
X_i	Gamma Air Dose Factor Gamma air dose factor for a uniform semi-infinite cloud of the radionuclide i. Values are found in Table 7.1-13.	(mrem/yr per $\mu Ci/m^3$)
$\overline{L}_i$	Beta Skin Dose Factor Beta skin dose factor for a semi-infinite cloud of the radionuclide i. Values are found in Table 7.1-13.	(mrem/yr per $\mu Ci/m^3$)

9.4.3.3 Inhalation Dose from the Nonnoble Gases

The inhalation dose rate and dose from the radioiodines, "particulates," and other (nonnoble gas) radionuclides during lake effect conditions are computed using Equations 9.28 and 9.29, respectively, except that χ/Q is used in lieu of $(1/u_s)\ (u\chi/Q)_s$ or $(1/u_g)\ (u\chi/Q)_g$. The χ/Q values of Table 9.4-2, as adjusted by the σ_θ -dependent factors of Table 9.2-6, are used in the calculations.

9.4.4 Symbols Used in Section 9.4

SYMBOL	NAME	UNIT
$D_{\gamma}(R)$	Whole Body Gamma Ray Dose	(mrem)
(χ/Q)	Relative Effluent Concentration	(sec/m^3)
A_i	Accumulative Release	(μCi)
$\bar{X}_i$	Whole Body Gamma Dose Factor	(mrem/yr per $\mu Ci/m^3$)
λ_i	Radiodecay Constant	(1/hr)
R	Downwind Range	(m)
u_g	Wind Speed, Ground Level	(m/sec)
$D_{skin}(R)$	Skin Dose at Distance R	(mrem)
X_i	Gamma Air Dose Factor	(mrem/yr per $\mu Ci/m^3$)
$\bar{L}_i$	Beta Skin Dose Factor	(mrem/yr per $\mu Ci/m^3$)

9.4.5 Constants Used in Section 9.4

NUMERICAL VALUE	NAME	UNIT
3.17×10^{-8}	Conversion Constant	(years/second)
3600	Conversion Constant	(sec/hr)

TABLE 9.4-1

ESTIMATED FREQUENCIES OF OCCURRENCE*

(Hours Per Year - Percent)

ZION STATION

NO LAKE EFFECTS	90
LAKE EFFECTS	10
	100
LAKE EFFECT TRAPPING	9
LAKE EFFECT - FUMIGATION	1
	10

*Based on 1978-1979 Hourly Measurements (March through September)

TABLE 9.4-2

RELATIVE CONCENTRATION (X/Q) (sec/m^3)

		RANGE (meters)					
		402	804	1609	3218	8045	16090
Case 1.	Stability A	4.6E-5	1.6E-5	5.2E-6	2.4E-6	7.2E-7	2.8E-7
	Stability B	4.4E-5	2.6E-5	8.9E-6	3.5E-6	1.0E-6	3.9E-7
	Stability C	2.5E-5	4.2E-5	1.8E-5	5.9E-6	1.5E-6	5.6E-7
	Stability D	5.5E-7	1.8E-5	3.1E-5	1.7E-5	5.2E-6	2.0E-6
	Stability E	3.4E-12	1.8E-6	2.2E-5	2.6E-5	1.0E-5	4.7E-6
	Stability F	9.6E-23	1.6E-9	2.4E-6	1.3E-5	1.5E-5	8.0E-6
	Stability G	4.8E-81	1.1E-17	1.3E-9	5.5E-7	5.3E-6	7.1E-6
Case 2.	(All Stabilities)	3.5E-4	2.1E-4	1.1E-4	5.3E-5	1.8E-5	7.0E-6

Note: Maximum X/Q values for each range have been underscored.

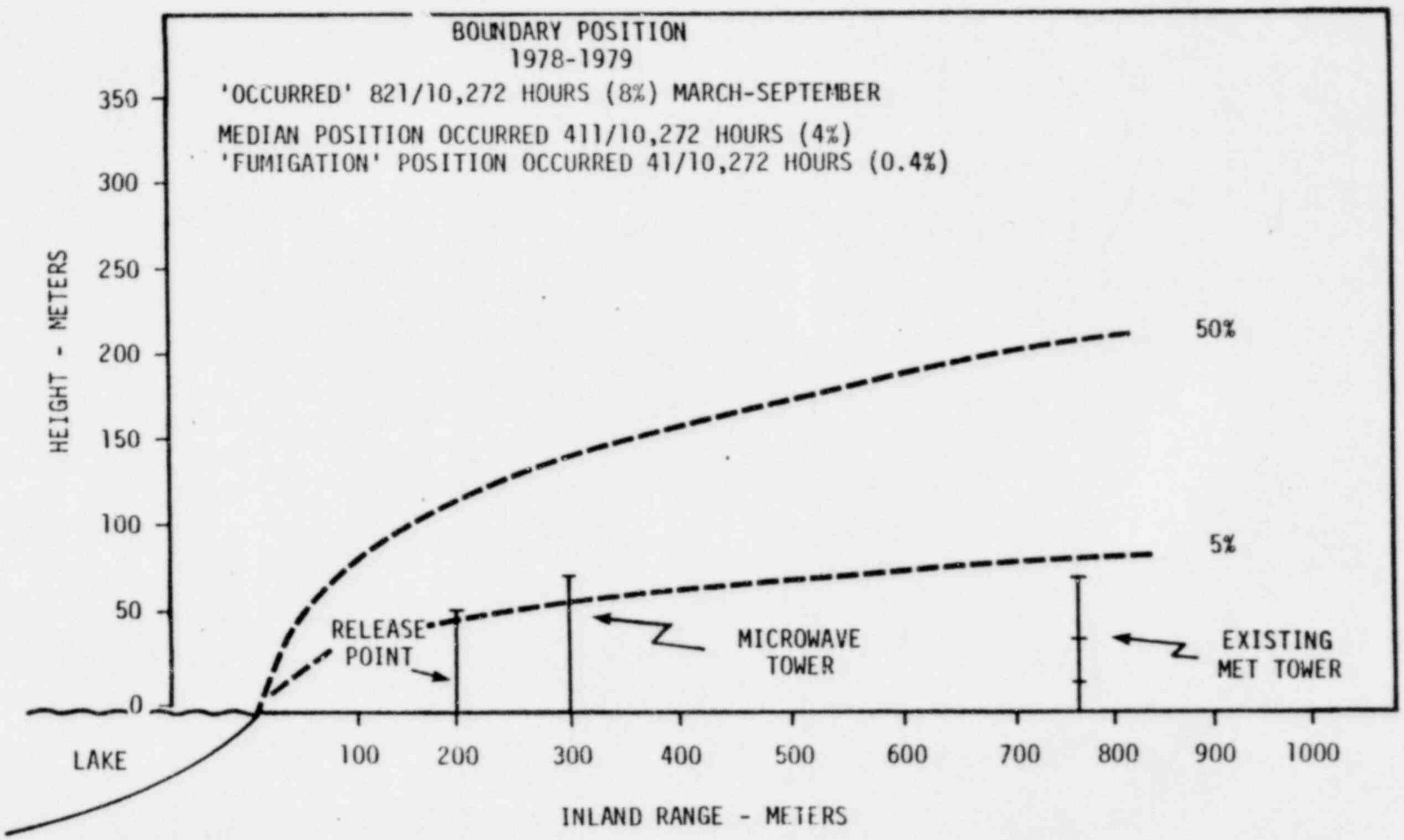
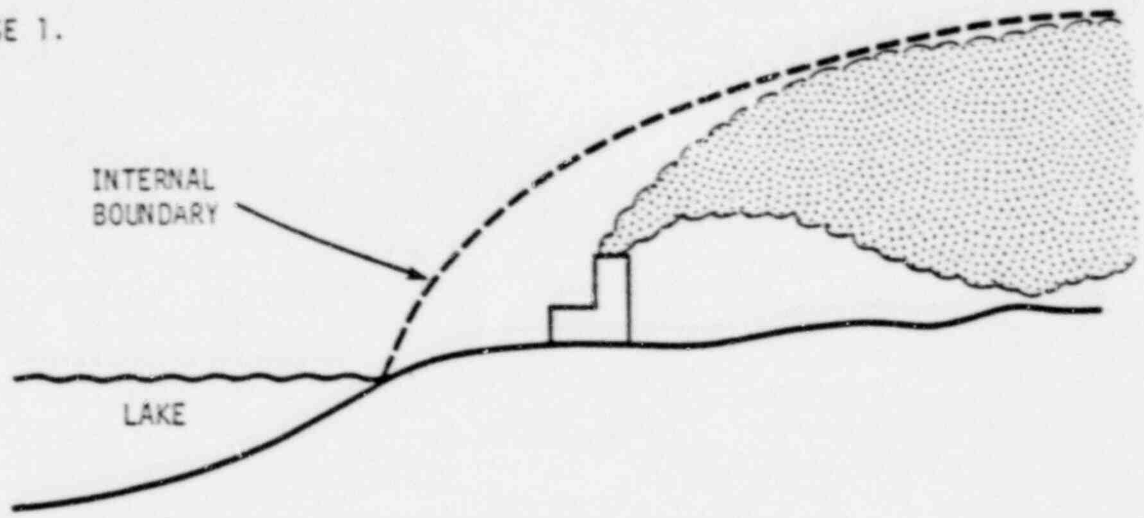


FIGURE 9.4-1

BOUNDARY POSITION

CASE 1.



CASE 2.

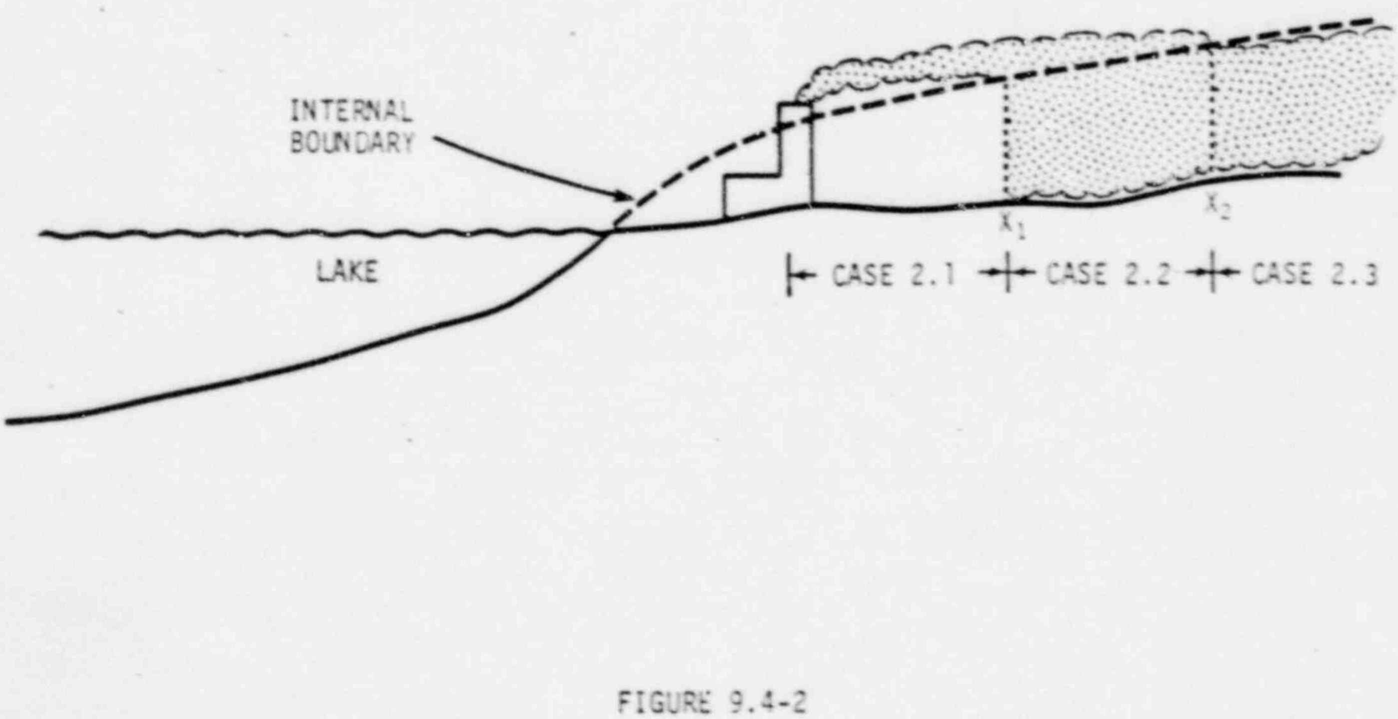


FIGURE 9.4-2

ATMOSPHERIC DISPERSION WITH LAKE EFFECTS

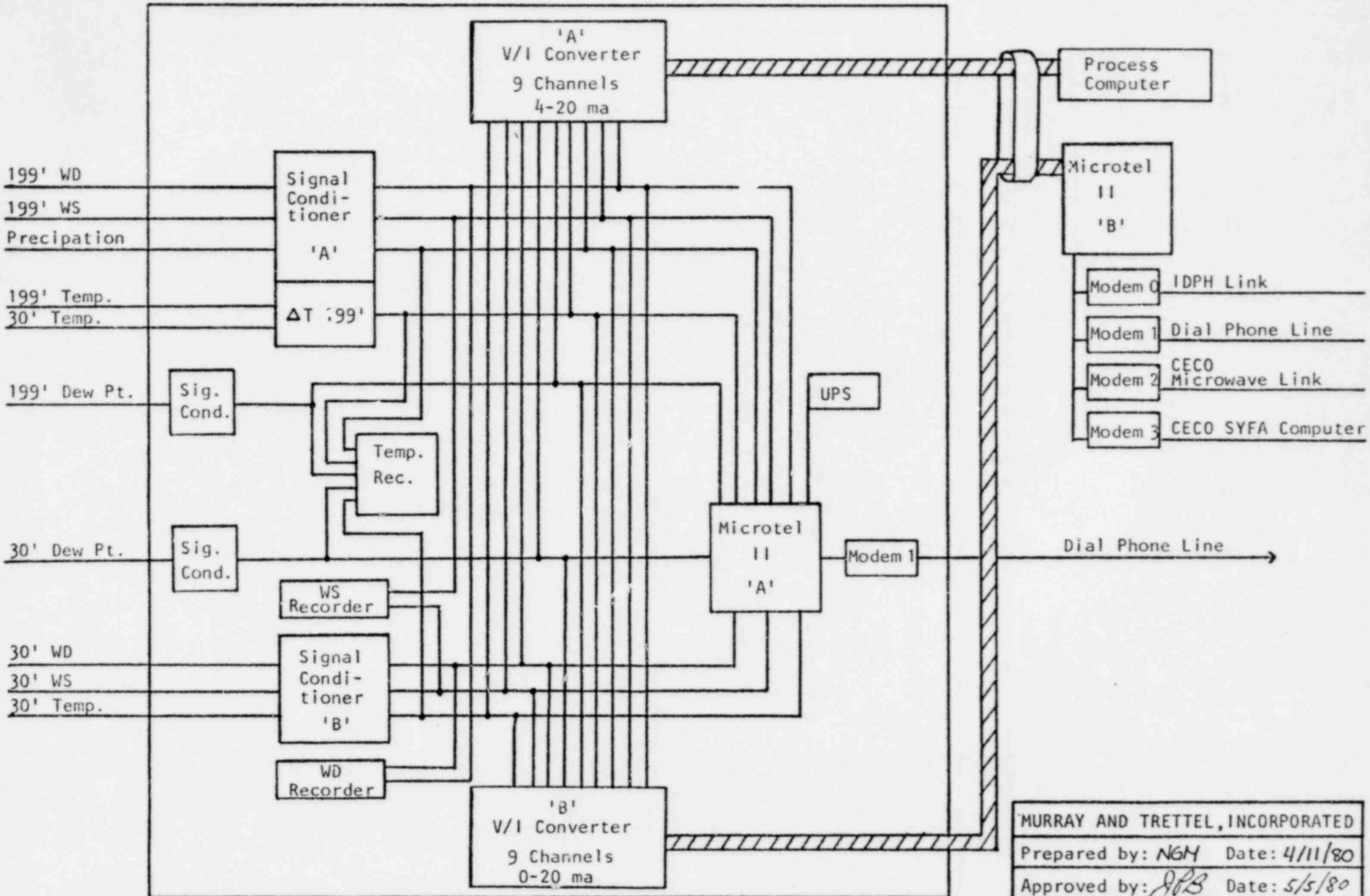
Appendix C

Independent Signal Pathways at Meteorological Facilities

Rev. [illegible]/5/80

BRAIDWOOD STATION

0-5V DC
Signal Conditioner Outputs



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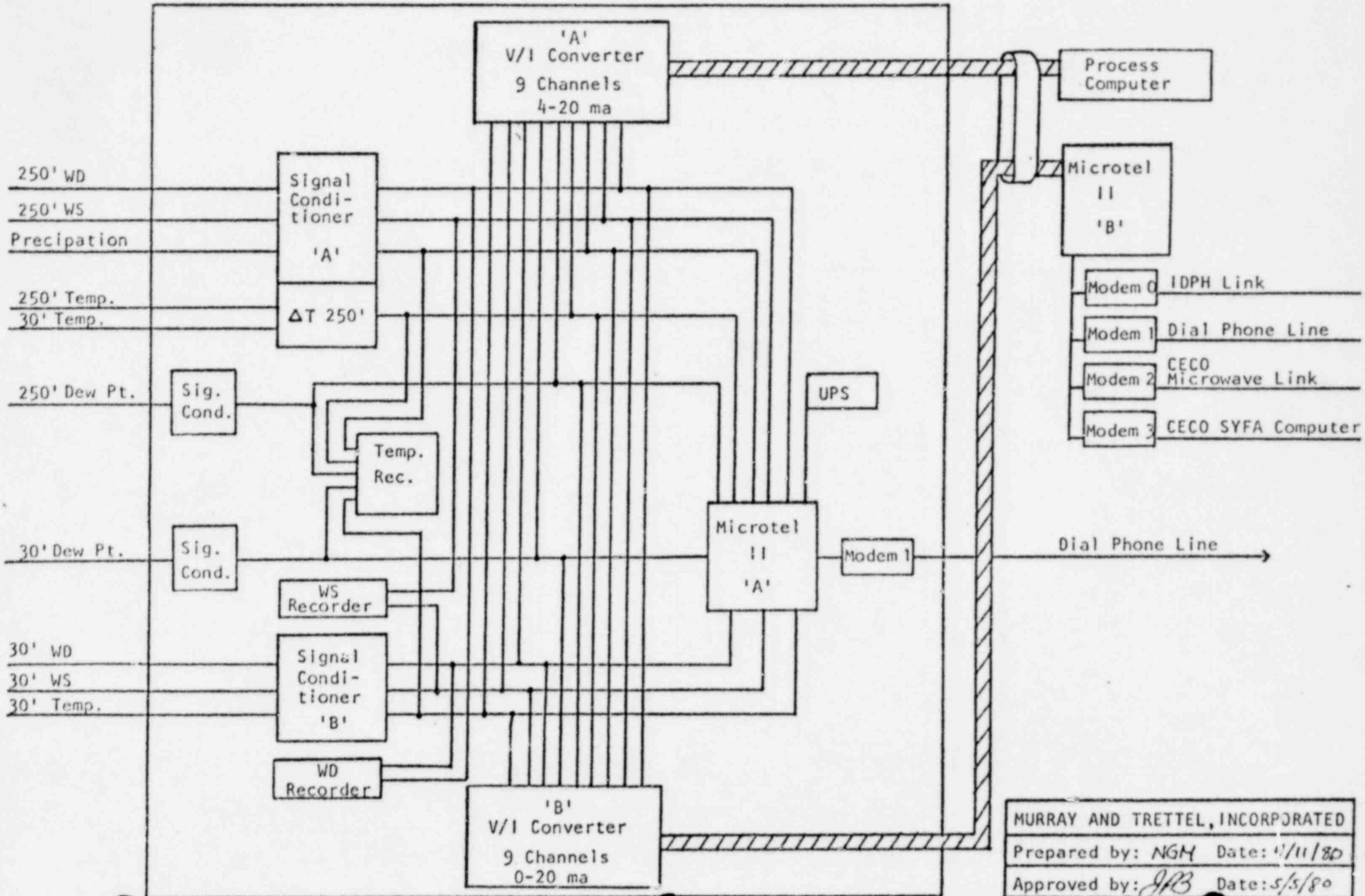
Prepared by: NGM	Date: 4/11/80
Approved by: JPB	Date: 5/5/80

Rev. 1 5/5/80

BYRON STATION

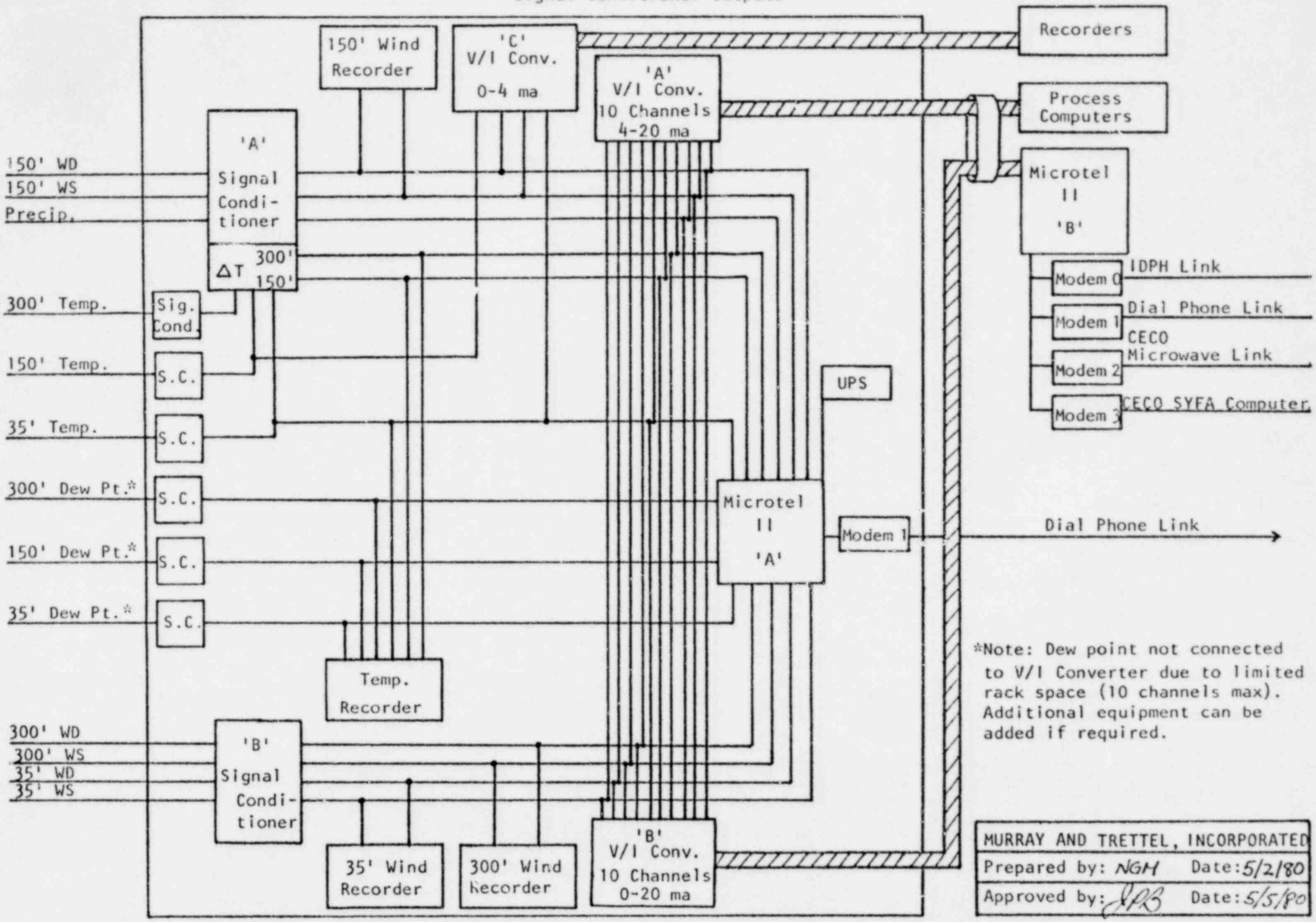
0-5V DC
Signal Conditioner Outputs



MURRAY AND TRETTEL, INCORPORATED	
Prepared by: NGM	Date: 4/11/80
Approved by: JRB	Date: 5/5/80

DRESDEN STATION

0-5V DC
Signal Conditioner Outputs



*Note: Dew point not connected to V/I Converter due to limited rack space (10 channels max). Additional equipment can be added if required.

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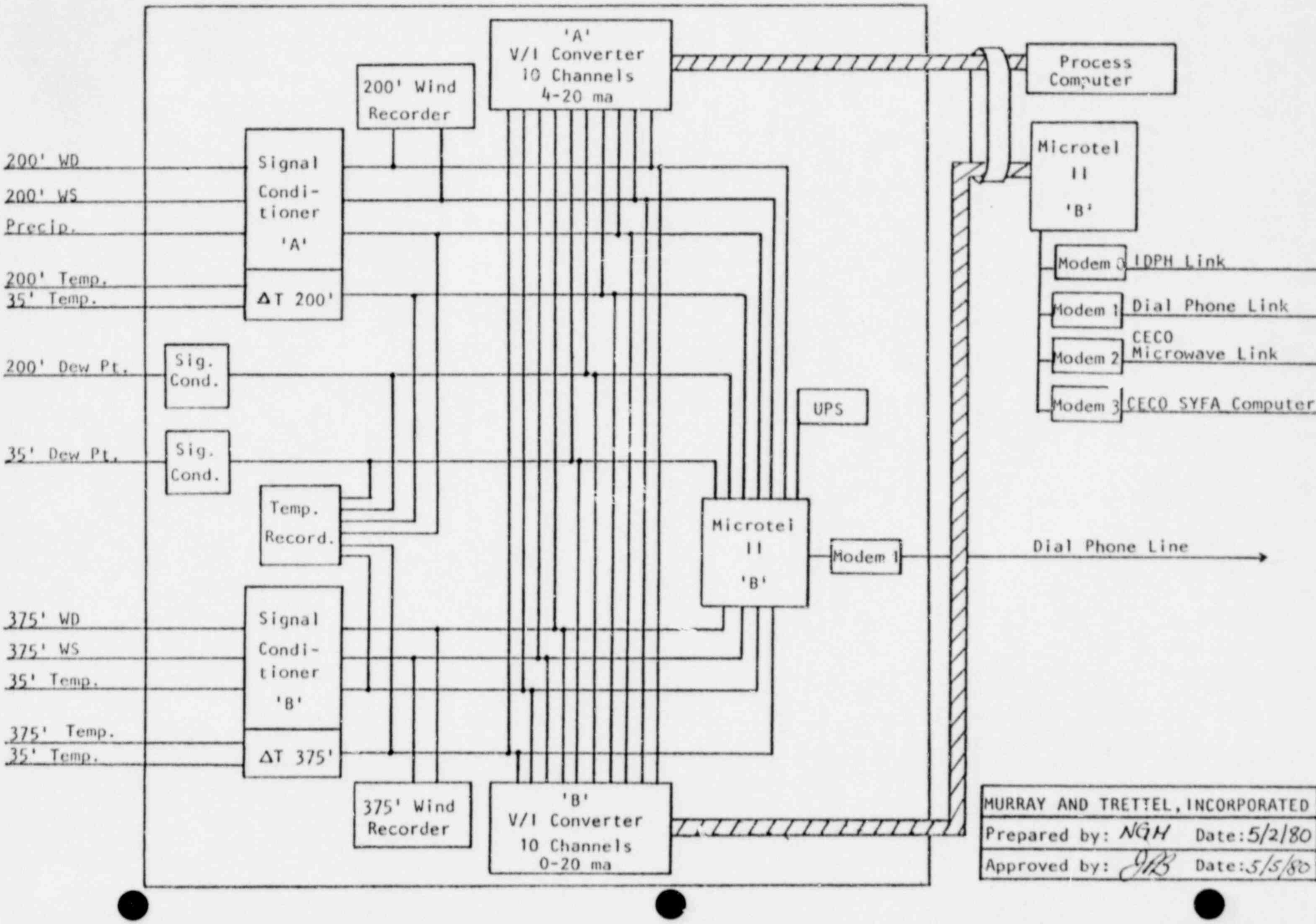
Approved by: JPB	Date: 5/5/80

Rev. 1 5/5/80

LASALLE STATION

0-5V DC
Signal Conditioner Outputs

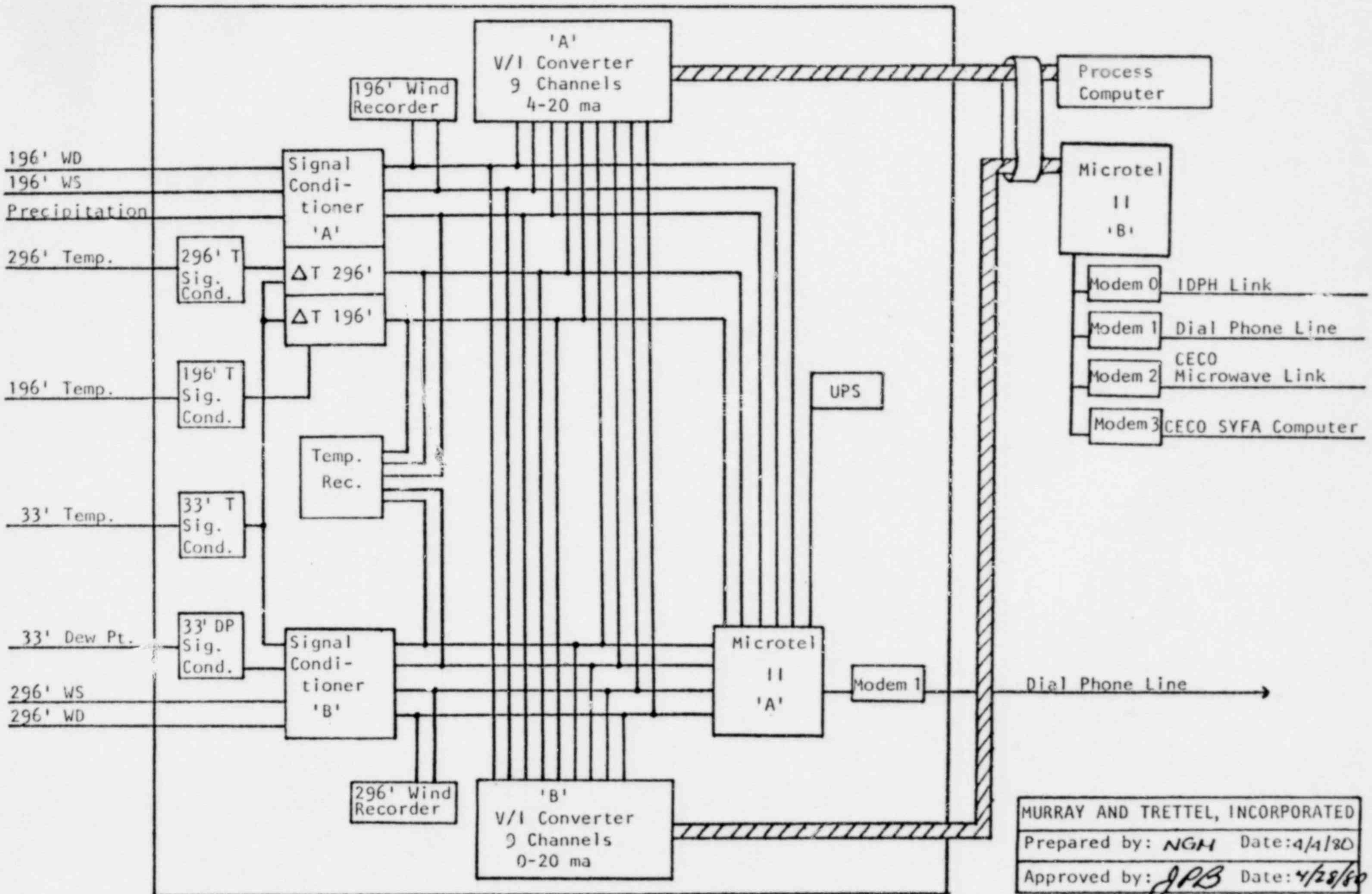


MURRAY AND TRETTEL, INCORPORATED	
Prepared by: NGH	Date: 5/2/80
Approved by: JRB	Date: 5/5/80

Rev. 1 5/5/80

SOUTH QUAD STATION

0-IV DC
Signal Conditioner Outputs



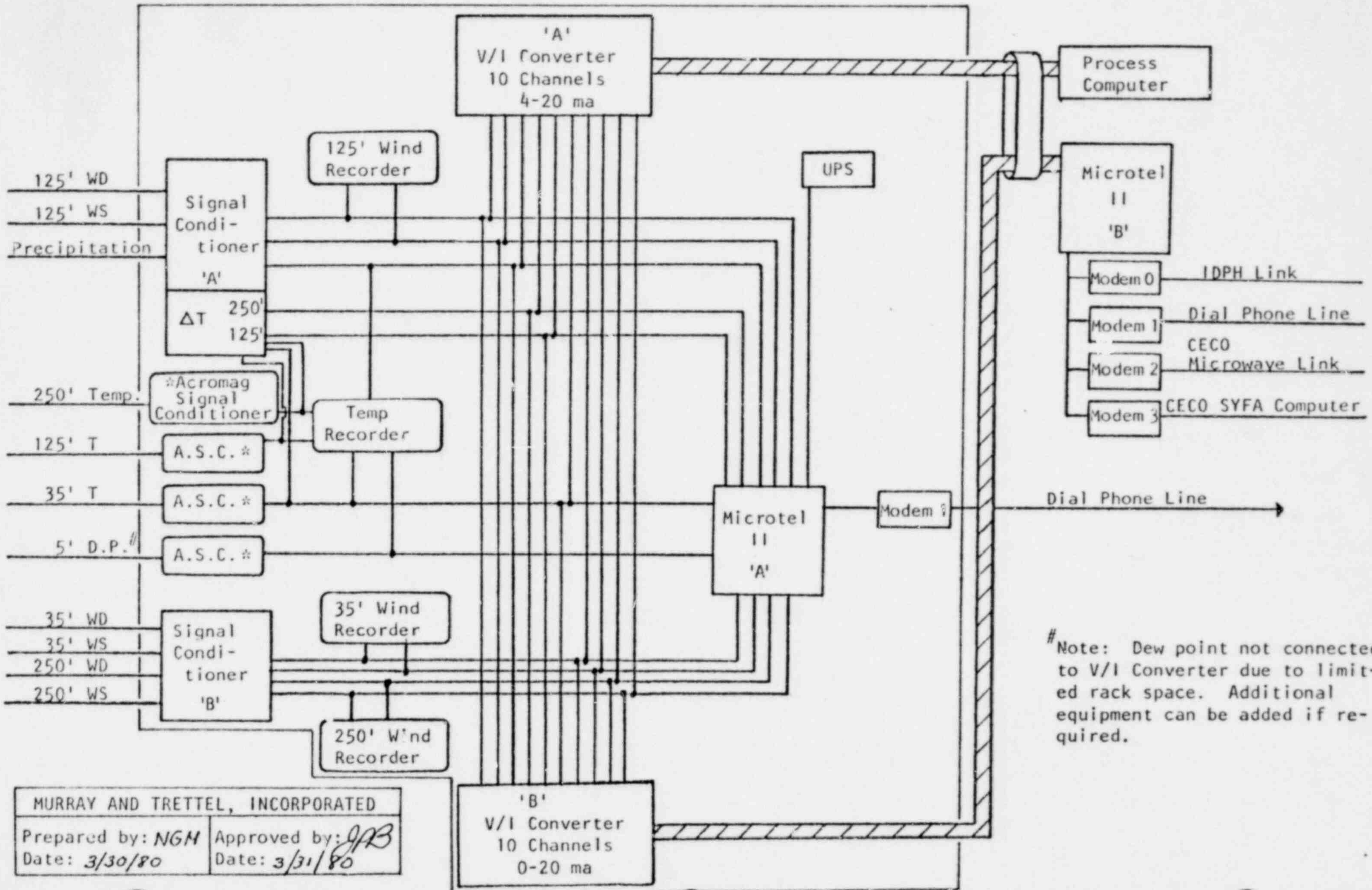
MURRAY AND TRETTEL, INCORPORATED	
Prepared by: NGH	Date: 4/4/80
Approved by: JPB	Date: 4/28/80

Rev. I 5/5/80

ZION STATION

0-1V DC

Signal Conditioner Outputs



#Note: Dew point not connected to V/I Converter due to limited rack space. Additional equipment can be added if required.

MURRAY AND TRETTEL, INCORPORATED	
Prepared by: NGM Date: 3/30/80	Approved by: JAB Date: 3/31/80

Appendix D

Study of Low-Level Dispersion Over Northern Illinois

Study of Low-Level Dispersion

Over Northern Illinois

prepared for

COMMONWEALTH EDISON COMPANY
Chicago, Illinois 60690

by

MURRAY AND TRETTEL, INCORPORATED

June 1974

ABSTRACT

This report describes a meteorological program to be conducted by Commonwealth Edison Company (CECO) at six nuclear station sites in northern Illinois during 1974. The object of this investigation is to demonstrate that a consolidation of the CECO meteorological programs is feasible. If sufficient justification is shown, the requirement for a fully instrumented tower at each CECO nuclear power plant site in accordance with Regulatory Guide 1.23 will be waived and appropriate data from the consolidated program will be permitted to describe the meteorological conditions at each site.

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1. Introduction

Starting 1 January 1974, Commonwealth Edison Company (CECO) plans on simultaneously operating meteorological monitoring programs at six (6) nuclear generating station sites located in northern Illinois: (1) Dresden (DNPS), (2) QUAD-Cities (QUAD), (3) Zion (ZION), (4) Braidwood (BRWD), (5) Byron (BYRN), and (6) one undocketed nuclear site Carroll County (CLCY). The maximum distance between any two locations is about 80 miles; the minimum distance is 12 miles.

The meteorological measurements processed through these monitoring programs afford a unique opportunity to investigate the lower 150 meters of the atmosphere over the northern portion of Illinois with respect to the dispersion of effluents from chimneys, vent stacks and cooling towers.

CECO believes that maintaining the meteorological monitoring programs as they presently exist may be shown to be unnecessary. This possibility has developed, in part, from the observations (1) that the differences between semi-annual relative concentration factors (X/Q) at the present sites are relatively small; (2) that there are apparently some strong relationships between the meteorological measurements made at the various sites; and, (3) that some of the current measurements can be inferred from other measurements either singly or in combination.

The concepts implied by the above statements as they relate to possibie revisions to the present policy of maintaining meteorological monitoring programs were presented by CECO in a letter from Mr. Byron Lee to Mr. J. F. O'Leary, Director, Directorate of Licensing, U.S. Atomic E ergy Commission dated 20 August 1973.

This report has been prepared at the request of Commonwealth Edison Company after they received a favorable response to their letter from Mr. A. Giambusso, Deputy Director for Reactor Projects, Directorate of Licensing, USAEC. The study is a joint effort by Commonwealth Edison Company and by Murray and Trettel, Inc., Certified Consulting Meteorologists.

There are five major objectives to the study:

Objective 1. To develop a computerized data bank consisting of hourly values of wind speed, direction, temperature and humidity based on instrumentation mounted on towers located at six CECO nuclear generation sites; incorporate hourly observations taken at five National Weather Service (NWS) installations located in northern Illinois.

Objective 2. To compare 1974 CECO meteorological data statistics from site to site and from level to level at each site to determine if useful correlations exist.

Objective 3. To study CECO and NWS data to determine how synoptic conditions influence the correlation of the 1974 CECO data so that transformations may be made to improve the correlations.

Objective 4. To analyze the 20-year record of Argonne National Laboratory (ANL) meteorological measurements to ascertain year to year variability of site boundary, maximum annual and short-term* average ground-level air concentrations, and compare these data to the 1974 CECO site records in order to show that the short-term station to station variability is consistent with the long-term single station variability of concentrations.

Objective 5. To prepare a humidity climatology for those sites equipped with tower-mounted humidity sensing devices in order to develop fog frequency statistics.

The objectives as described above evolved, in part, from a small pilot study which has been included as part of this report. The analytical techniques and methodologies used in the pilot study, although typical of those anticipated in the detailed plan, should be considered as initial and tentative, subject to change if further investigation warrants.

Commonwealth Edison Company is cognizant of the present Atomic Energy Commission requirement of one year of on-site measurements prior to operation. This requirement will be met unless the results of this study indicate otherwise, and, then, only with the concurrence of the Atomic Energy Commission.

* NOTE: Eight and sixteen hour plus three and twenty-six day time periods.

II. Scope

The locations of the CECO Nuclear Power Sites, the ANL reactor site and five NWS stations are depicted in Figure 1. The weather stations are Moline (MLI), Rockford (RFD), DuPage County Airport (DPA), O'Hare Airport (ORD) and Midway Airport (MDW). The sites are clustered in the northernmost portion of Illinois where the terrain can be described generally as gently rolling prairie with only minor variations in topography. Two of the CECO sites can be considered to have unique topographical features that might influence their on-site meteorology or climatology, and thereby weaken any correlations with the other sites. Zion Station, located on the shore of Lake Michigan, is subjected to intrusions of air modified by the lake, and the Carroll County Station Site, situated atop a bluff along the east bank of the Mississippi River, has topographical variations that are more pronounced than elsewhere.

Site elevations (msl) and tower heights (above grade) are summarized in Table 1. The lowest site elevation is at DNPS (514 ft.) and the highest is at BYRN (840 ft.), a difference of 326 feet.

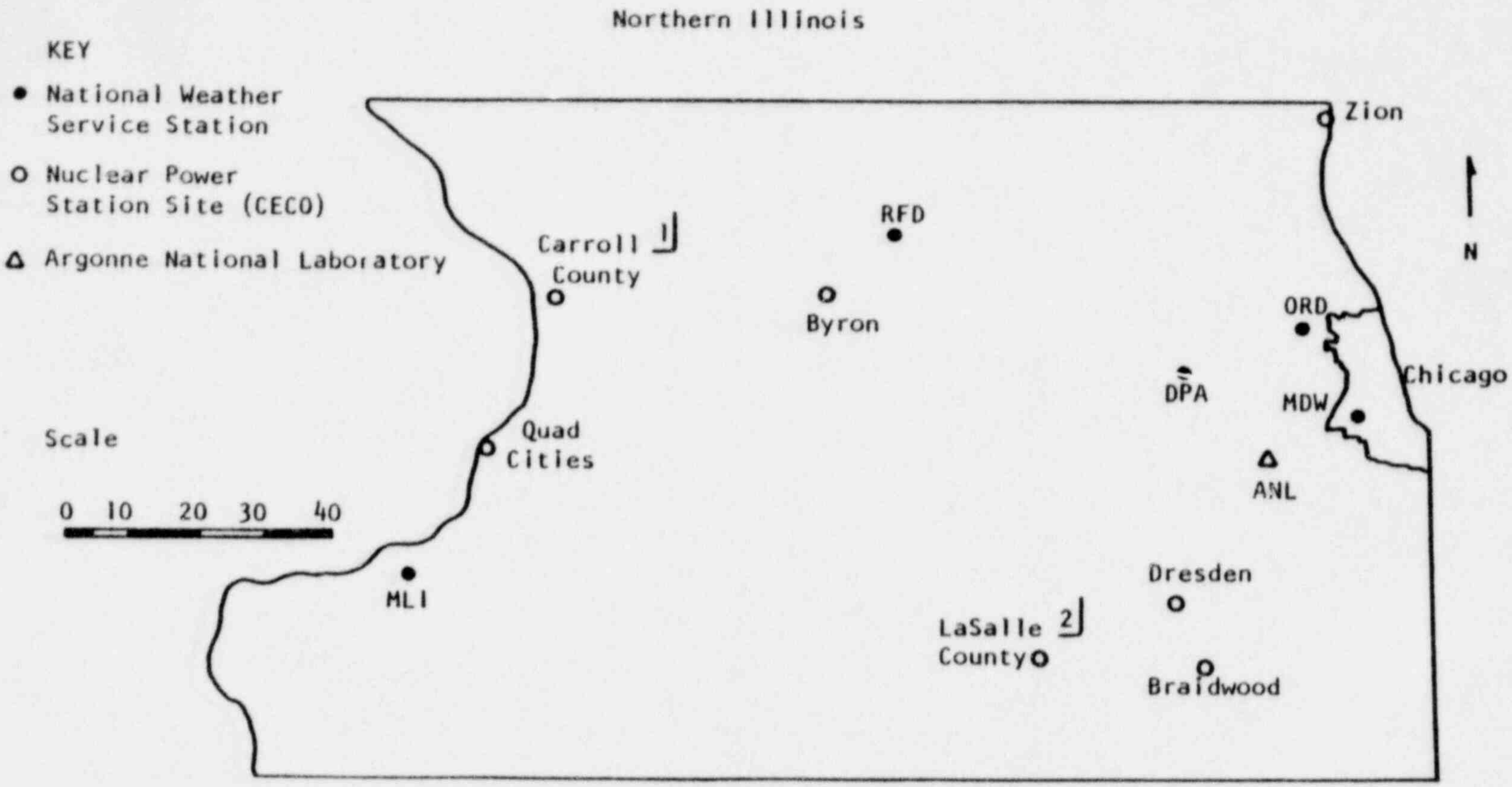


Figure 1. Location Map of CECO Nuclear Power Stations, National Weather Service Stations and Argonne National Laboratory

1 Undocketed nuclear site.

2 Meteorological measurements not available during 1974.

TABLE 1

Site Elevations and Meteorological Tower Heights

Site	Identifier	Elevation (ft., msl)	Tower Height (ft.)
Argonne	ANL	746	150
Braidwood	BRWD	600	199
Byron	BYRN	840	250
Carroll County	CLCY	720	499
Dresden	DNPS	514	300
LaSalle *	LSCS	710	375
Quad-Cities	QUAD	595	400
Zion	ZION	590	250

* LSCS measurements not available for this study.

Primary measurements of wind direction and speed, air temperatures, differential temperatures, dew point temperatures (or relative humidity) and precipitation will be available on an hourly basis from each of the above mentioned six CECO stations during 1974. Descriptions of meteorological measurements at each of the CECO sites are given in Table 2 , along with the elevation above grade that each sensing device is located.

At Zion and Quad wind speed and direction are measured with Belfort Type L wind transmitters located on tower-mounted platforms. Air temperature is measured with Bristol resistance bulbs housed in aspirated radiation shields also mounted on the platforms. Dew point temperature is measured with a Foxboro Dewcell element mounted in a standard instrument shelter. Precipitation is measured with a Belfort weighing type rain gauge equipped with an alter-type windshield.

Wind velocity data are recorded on Belfort Type M wind recorders. The data are represented on analog charts by continuous pen traces. The chart is separated into two sections, each 3.75 inches wide. The speed section is graduated every 2 mph over the range from 0 to 120 mph. The direction section is graduated every 5 degrees over a 360 degree range.

Air temperatures and dew point temperatures are recorded on a Westronics Model M1102 multipoint recorder.

The 11-inch chart is graduated every degree over a range from -40 F to +120 F as well as every 0.25 F over a range from -10 F to +30 F. Ambient air temperature and the dew point temperature are referenced to the first scale and differential

TABLE 2

Commonwealth Edison Meteorological Facilities

Measurement	BRWD	BYRN	CLCY	DNPS	LSCS *	QUAD	ZION
	Measurement Height Above Ground (feet)						
Wind Direction/Speed	30 199	30 250	33 170 300 499	35 150 300	33 375	35 125 300 400	35 125 250
Ambient Temperature	30	30	33	35	33	35	35
Differential Temperature	199-30	250-30	170-33 300-33 499-33	150-35 300-35	375-33	125-35 300-35 400-35	125-35 250-35
Dew Point Temperature	30 199	30 250	33 499	35 150 300	33 375	5	5
Precipitation	2	2				2	2

* LSCS measurements not available for this study.

air temperatures to the second. Precipitation amounts are indicated by a continuous pen trace on a six-inch wide chart graduated every 0.05 inches over a range from 0 to 12 inches.

At DNPS wind speed and direction are measured with Teledyne Geotech Series 50 sensors located on tower-mounted booms. Air Temperatures (dry bulb and dew point) are measured with Cambridge Systems Model 1105-M Automatic Meteorological Temperature and Dew Point Measuring sets.

Wind velocity data are recorded on Esterline Angus Series E rectillinear recorders. The chart is separated into two sections, each 4.50 inches wide. The speed section is graduated every 2 mph over the range from 0 to 100 mph. The direction section is graduated every 10 degrees over a 540 degree range. Temperatures are recorded on a Bristol Model 550 multipoint recorder. The 11 inch chart is gradrated every 2 degrees over a range from -80F to +120F.

Voltage equivalents of all measurements are recorded on a magnetic tape acquisition system. Wind data are scanned every ten seconds and temperature data each minute.

The BYRN, BRWD and CLCY wind systems are MRI model 1074-1 combined cup and vane sensors. Temperature and humidity sensors at BYRN and BRWD are MRI model 800 series aspirated sensors. At CLCY, Cambridge Systems model 1105-M dew point sets will be used in conjunction with MRI model 811 Temperature Sensors.

The sensing threshold of the Belfort wind equipment is approximately 2 mph; that of the Teledyne equipment is 0.9 mph and that of the MRI equipment is 0.5 mph. Accuracy of the Cambridge systems equipment is 0.4 F; that of the MRI equipment is 0.8 F, and that of the Bristol equipment is 0.8 F.

Field tests and calibration procedures are performed monthly at BYRN and BRWD, every other month at DNPS, CLCY, QUAD and ZION. The following list of procedures is typical of those used at the sites:

A. Routine Maintenance Procedures and Equipment Check List

Inventory

Test Connectors	Sandpaper
Nuts and Bolts	Taps and Dies
Dew Point Calculator	Wet Bulb Wicks
Equipment Manuals	Extension Cord
5 Gallons Water	Compass
Pen Cleaning Wires	Li Cl Solution
Pen Syringe	Non Gumming Instrument Oil
9-20 VDC Power Supply	Spare Parts - Tubes
1-inch Rainfall Calibrator	Pens, Inked Felts, Motor
Digital Multi-Meter	Spray Cleaners
Walkie Talkies	Fuses
Tool Kit	Silicone Dashpot Oil
Towels	Test Spools (RTD)

Procedures

1. When working on recorders, mark charts when correcting time scale or moving pens so chart readers will not misread chart.

2. Check the syncro "zero" on all wind recorders (remove from case) and adjust as necessary.

3. Check pens for proper movement and span and direction pens and adjust as needed.

4. Check wind speed pen "zero" and adjust as needed. Apply full scale voltage to wind speed input and check accuracy of movement.

5. Lubricate recorders according to instruction manual.

6. Remove, clean and flush pens with ink as required.

7. Have tower-man position wind transmitter to appropriate marker and check for proper indication on recorder; adjust as needed.

8. Have tower-man rotate wind direction transmitter and check it for freedom of movement. Check recorder response.

9. Tower-man will replace all temperature bulbs and bulb covers as needed.

10. Mark recorder charts and check chart speed.

11. Replace inked felts in multipoint recorder.

12. Spray Contact Re-Nu on slidewires, stepping switches, with power off and rotate slidewire to wipe.

13. Check resistance on shorted set of resistance thermometer leads.

14. Disconnect each resistance thermometer and check for proper resistance according to temperature table in manual (subtract resistance on shorted set to obtain true reading).

15. Check resistance thermometer leads to ground for leakage.

16. Lubricate recorder motors and gears according to manual.

17. Check temperature recorder for proper "deadband" and adjust gain as necessary.

18. Check recorder calibration against resistance data above and adjust as needed.

19. Wash Dewcell element with water and test for cleanliness according to manual. Re-salt element with dilute solution of Lithium Chloride.

20. Verify thermometer and sling psychrometer are present and record temperatures. Check temperatures (dry bulb and dew point) recorder values.

21. Remove rain gage cover, empty bucket, and clean.

22. Clean weighing mechanism with spray cleaner, then lubricate with non-gumming instrument oil according to manual.

23. Fill Dashpot with silicone oil as needed.

24. Check rain gage "zero" with empty bucket in place an adjust as necessary.

25. Add water to bucket and check calibration; adjust as necessary. Empty bucket.

Temperature elements are periodically tested by immersing them in ice baths and comparing recorder values for proper temperature.

In addition, hourly sequence reports for Midway Airport (MDW), O'Hare (ORD), DuPage County (DPA), Rockford (RFD) and Moline (MLI), including surface wind and temperature, atmospheric pressure, and cloud types and heights on an hourly basis will be available and used when necessary.

The data will be processed using a CDC 6000 Series Computer. The information will be updated and edited from time to time during the year and stored on magnetic tape for subsequent use. The quantity of information to be processed makes the creation of a data bank necessary.

III. Analytical Procedures

Part One:

Data Bank

The measurements currently being taken at the CECO stations are analyzed on an hourly basis during the routine course of the monitoring programs. These hourly station data, when processed for the purposes of the monitoring programs, will be merged together and written on magnetic tape.

NWS data are routinely transmitted to Murray and Trettel and printed by teletype. These reports will be coded to indicate the altitude and amount of low, middle and high clouds, visibility, weather and obstruction to vision, sea-level pressure, temperature, dew point and wind, along with the station identifier and time. These data will be entered directly by remote terminal to the CDC 6000 series computer.

The data bank will be structured to permit editing and up-dating as needed.

Data collected at the CECO stations are summarized in Table 2. A partial description of the ANL insturments is given in Table 3 taken from their report ANL-7084*; a more complete and updated (to 1971) list of the measurements available on magnetic tape is given in Table 4, also taken from ANL-7084.

* NOTE: Fifteen-year climatological summary, January 1, 1950 - December 31, 1964 by Harry Moses and Mary A. Bogner, ANL-7084, September 1967.

TABLE 3

Argonne National Laboratory Meteorological Instrumentation

Meteorological Element	Type of Instrument
Wind Speed	Friez Aerovane
	Belfort 3-cup Anemometer Model No. 339-A
Wind Direction	Friez Aerovane
Temperature	Friez Hygrothermograph Model No. 594
	5-junction Copper-constantan Thermopile No. 36 wire and Honeywell Electronik Potentiometer
	5-junction Copper-constantan Thermopile No. 16 wire and Honeywell Electronik Potentiometer
Dew Point Temperature	Foxboro Dewcel
Relative Humidity	Friez Hygrothermograph Model No. 594
Stability	Copper-constantan thermocouple No. 36 wire at all levels and Honeywell Electronik Potentiometer
Precipitation	Friez Weighing-type Rain Gage Model No. 755-B
	Bendix Friez Tipping Bucket Precipitation Gage
Direct and Diffuse Solar Radiation	Eppley 50-junction Pyranometer and Leeds and Northrup Micromax Recorder
Net Radiation	Beckman & Whitley Net Radiometer Model No. N-188

Meteorological Element	Type of Instrument
Pressure	Friez Microbarograph Model No. 790-1
Soil Temperature	Leeds and Northrup 100-ohm copper thermohms and Micromax Recorder

TABLE 4

ARGONNE NATIONAL LABORATORY

Hourly Weather Data Format

Card Image Column	Information
1-2	blank
3-5	card identifier (103)
6-7	last two digits of year
8-9	month (01-12)
10-11	day of month (01-31)
12-13	hour of day (01-24), Central Standard Time
14	sign column (see below)
15-17	1.17 foot dewpoint (F)
18	sign column
19-21	2.34 foot dewpoint (F)
22	sign column
23-25	4.69 foot dewpoint (F)
26	sign column
27-29	9.38 foot dewpoint (F)
30	sign column
31-33	131 foot dewpoint (F)
34	sign column
35-37	temperature difference (F), 15.2 minus 5.5 feet
38	sign column
39-41	temperature difference (F), 34.0 minus 5.5 feet *
42	sign column
43-45	temperature difference (F), 72.2 minus 5.5 feet
46	sign column
47-49	temperature difference (F), 144 minus 5.5 feet *
50	sign column
51-53	net radiation - total Langleys per hour
54	blank
55-57	solar radiation - total Langleys per hour
58	blank for no precipitation, T for trace of precipitation
59-61	precipitation in hundredths of inches
62	sign column
63-66	ambient air temperature (F) at 5.5 feet
67	blank
68-71	station pressure in inches of mercury
72	blank
73-75	relative humidity (percent)

Code for sign columns:

Blank - Entry immediately following is positive.

Negative sign - Entry immediately following is negative.

* NOTE: ANL temperature differential from 144-ft. to 34-ft. will be derived from (144-5.5) -(34.0-5.5) differences.

TABLE 4 (continued)

ARGONNE NATIONAL LABORATORY

Hourly Weather Data Format

Card Image Column	Information
76-77	blank
78-80	card identifier (141)
81-82	last two digits of year
83-84	month (01-12)
85-86	day of month (01-31)
87-88	hour of day (01-24), Central Standard Time
89	blank
90-92	hourly wind speed at 9.38 feet in miles and tenths per hour
93	blank
94-96	hourly wind speed at 18.75 feet in miles and tenths per hour
97	blank
98-100	hourly wind speed at 37.5 feet in miles and tenths per hour
101	blank
102-104	hourly wind speed at 75 feet in miles and tenths per hour
105	blank
106-108	hourly wind speed at 150 feet in miles and tenths per hour
109	blank
110-111	wind direction at 18.75 feet in tens of degrees (01-36)*
112	blank
113-114	wind direction at 150 feet in tens of degrees (01-36)*

* Prior to October 1960, a wind direction recorded as 37 is variable and 00 is calm.

Part Two:

Correlations Among CECO Meteorological Measurements

Multiple regression analysis techniques will be used to determine the degree of correlation among the measured variables at the different on-site tower elevations, as well as between sites.

The regression programs and correlation analyses will be taken from an existing package of computer programs called the Statistical Package for the Social Sciences * (SPSS). The regression program provides any of three methods of multiple regression with variable selection: forward, stepwise, or backward elimination.

For example: if the stability conditions at Site "x" can be inferred from one or more of the other variables measured at one or more CECO sites, then the measurement at Site "x" is redundant. Or, if the upper level wind direction and speed at elevation "x" can be inferred from the other measurements then it is redundant at elevation "x".

* Statistical Package for the Social Sciences (SPSS).Norman H. Nie, Dale H. Brent. C. Hadlai Hull, 1970. McGraw-Hill Book Company.

Part Three

Data Transformations

Synoptic influences may affect the correlation of data among towers or among levels of a tower causing a large scattering of the data. By documenting the synoptic conditions with the NWS data, occurrences of large scattering of CECO data might be explained by differences in synoptic conditions between sites and the correlations might be improved through data transformations using the NWS data. Pressure, cloud cover and visibility observations are included in the NWS data. These data parameters are not available from the CECO towers. The cloud cover information can be used to determine stability conditions as outlined by Turner*.

The purpose of this phase of the study is to determine how these data transformations can minimize the number and type of measurements required at each specific station. As an example, the results of part two may show that the 300 ft. to 35 ft. differential temperature can be related to the 125 ft. to 35 ft. differential temperature and the 35 ft. wind speed. In part three of the study it may be found that the addition of the sky cover observed at a nearby NWS station may significantly improve the correlation to the point where an actual measurement of temperature at the 300 ft. level becomes unnecessary.

Stepwise regression analysis will be employed to allow the selected variables to be introduced into the computation sequentially.

* A Diffusion Model for an Urban Area, 1964. D. Bruce Turner. Journal of Applied Meteorology, Vol. 3, pp. 83-91.

Part Four:

Comparative Dispersion Climatology

A. Annual Average Dispersion Climatology

A twenty-year record (1951-71) of hourly measurements including wind direction, speed and differential temperatures has been compiled by the meteorology group at ANL and is available on magnetic tape. The data will be processed into annual stability wind roses. Stability will be defined in terms of differential temperatures measured at 144 ft. and 34 ft. (or 144 ft. and 19 ft. when 34 ft. data not available). Wind velocity measured at 155 ft. (the highest instrumented elevation) will be utilized to calculate the ground-level relative concentration from the chimney of a hypothetical nuclear generation station; wind velocity measured at 19 ft. will be used for vent stack and ground-level releases. A record spanning at least ten years is desired since the objective of this study is to estimate the range of relative concentration resulting from the year to year variation of the meteorological data. A station is hypothesized with certain fixed characteristics.

The pertinent characteristics will be:

Chimney height:	95 meters (312 ft.)
Chimney diameter	3.36 meters (11 ft.)
Heat content:	2.0×10^6 cal/sec
Discharge Velocity:	200,000 CFM
Vent Stack Height:	46 meters (150 ft.)
Site boundary:	circle with radius of 500 meters (1,640 ft.)

Relative concentration values will be calculated from equation 5.15 (Turner)* with

* Workbook of Atmospheric Dispersion Estimates, 1969. PHS. Pub. No. 999-AP-26.

plume rise evaluated according to Briggs* for chimney releases. A ground-level release will be assumed for the vent. The maximum annual values will be determined for each year irrespective of range or direction. They will be ranked and a cumulative frequency distribution will be plotted on probability paper (with normalizing transformations if necessary) from which the 50, 10, 5.0 and 1.0 percent values will be determined. In addition, X/Q versus range will be calculated for each of 16 directions for each year. Year to year variations for each direction will then be determined.

The annual average ground-level concentrations will be calculated for the hypothetical station using each of the CECO stations 1974 annual stability roses. These concentrations will be compared to the probability curve for ANL to indicate whether they represent "low", "average", or "high" concentration years. "Average" will be defined here as those concentrations falling within the interval from 16 to 84 percent inclusive (± 1 Standard Deviation); "low" corresponds to those falling in the interval less than 16 percent (less than -1 standard deviation) and "high" corresponds to those in the interval greater than 84 percent (greater than +1 standard deviation).

* Plume Rise, 1970. G. A. Briggs. AEC Critical Review Series

B. Short-Term Dispersion Climatology

The ANL data will also be used to generate 8, 16, 72 and 624 hour site boundary "worst case" statistics for the hypothetical station using the "window" model similar to that described by Woodard.* The ANL 19 ft. elevation wind data will be used to calculate hourly "X/Q" values resulting from an assumed ground-level point source. Cumulative frequency distributions will be determined for 16 directions for the available years of record. The envelope of the curves will then be used to indicate the probable "worst case" conditions that could be experienced by the station during its lifetime. The 35 ft. stability wind rose data generated for each of the six CECO stations over the 1974 period will also be subjected to the same analysis and compared to the ANL curve. Joint persistence of wind direction-speed and stability will also be computed. Reasonable upper-limits to both maximum annual average concentrations and short-term "worst case" concentrations for the CECO network should evolve from these comparisons.

* Probability Treatment of Atmospheric Dispersion for Dose Calculations. K. Woodard. Nuclear Technology, Vol. 12, November 1971.

Part Five

Humidity Climatology

Dew point temperatures and/or relative humidity are measured at various elevations at DNPS, BRWD, BYRN, and at CLCY. S e cooling lakes or cooling towers are or will eventually be operated at these sites, knowledge of the humidity distribution becomes important. Hourly values of relative humidity will be averaged over each month. The average, maximum and minimum values will be tabulated and plotted on a figure having relative humidity as ordinate and hour of day as abscissa. A composite figure showing relative humidity as a function of month and hour will be generated. These tables and figures will reveal graphically those periods of high and low relative humidity, both on the diurnal cycle (day versus night) and the seasonal cycle as well. The likelihood of natural fog and/or lowered visibility can be inferred from the humidity information.

The presence of fog implies high relative humidity, conversely high relative humidity measurements usually imply the strong possibility of fog. In the absence of direct fog measurements, the frequencies of relative humidity values equal to or greater than 90 percent will provide an estimate of the number of hours per year fog might have occurred. Moreover, the most likely times for fog (diurnal and seasonal variation) will be inferred. Persistence of humidity in excess of 89 percent will be generated for each month to give an indication of how many hours of fog is likely to persist (median value) as well as how long it can persist (maximum). The 50, 80, 90, 95, 99 and 99.9 percentiles will be generated to provide this information. Figures 2 and 3 and Table 5 typify the format of data presentation for this study.

The dew point temperatures will be converted to relative humidity by calculating the saturation vapor pressure for both the dry bulb (T_{db}) and dew point (T_{dp}) temperatures and taking their ratio. The saturation vapor pressure (e_s) will be computed according to the formula:*

$$\log e_s = 8.4051 - 2353/T \qquad (*)$$

Where e_s is in centibars and T is in Kelvin.

The relative humidity, therefore is given as

$$\text{Rel. Hum. } (\%) = \frac{10^{\,8.4051 - 2353/T_{dp}}}{10^{\,8.4051 - 2353/T_{db}}} \times 100$$

* Page 51; Dynamic Meteorology; Holmboe, Forsythe, Gustin. 1957

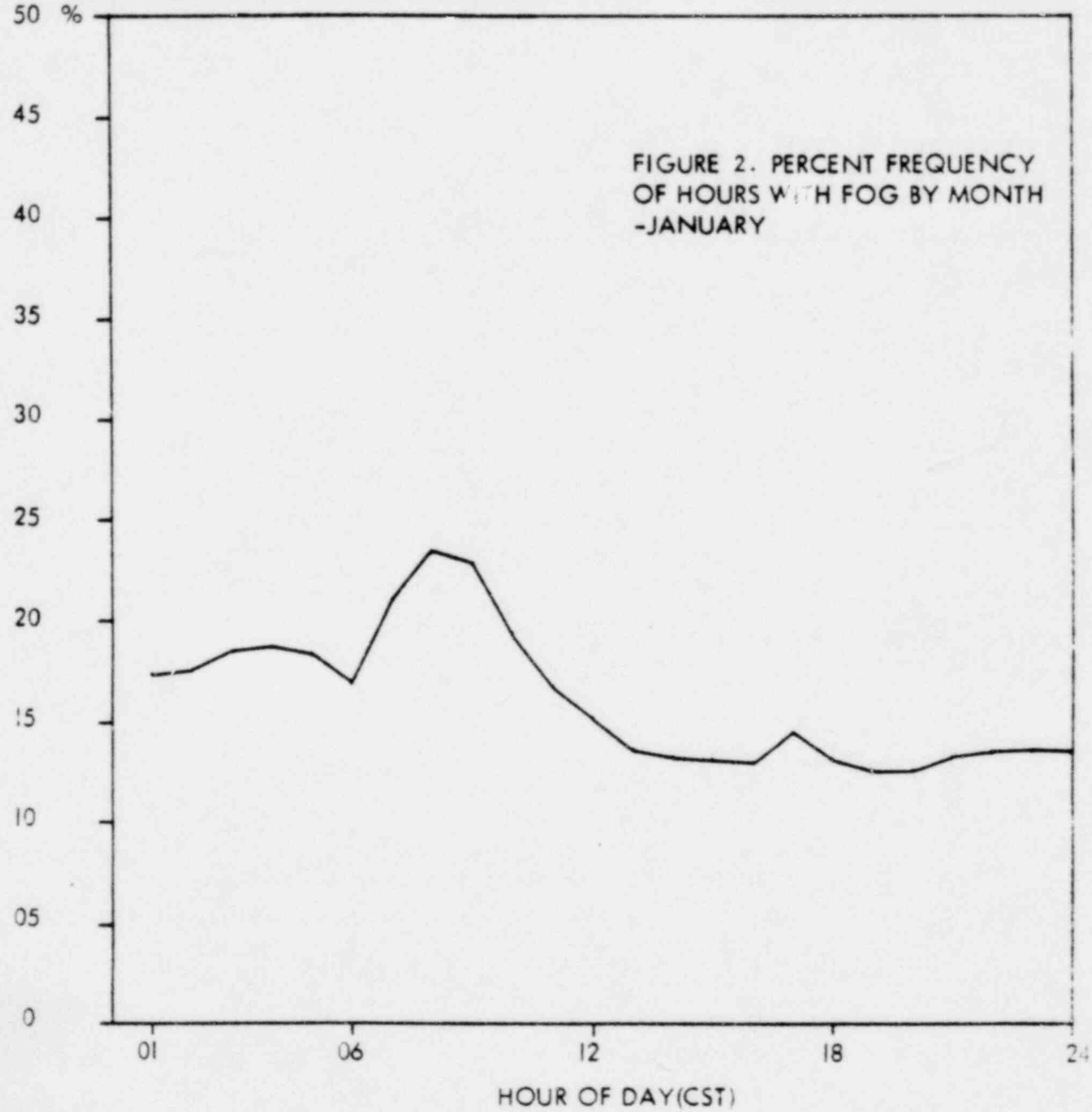


FIGURE 2. PERCENT FREQUENCY OF HOURS WITH FOG BY MONTH -JANUARY

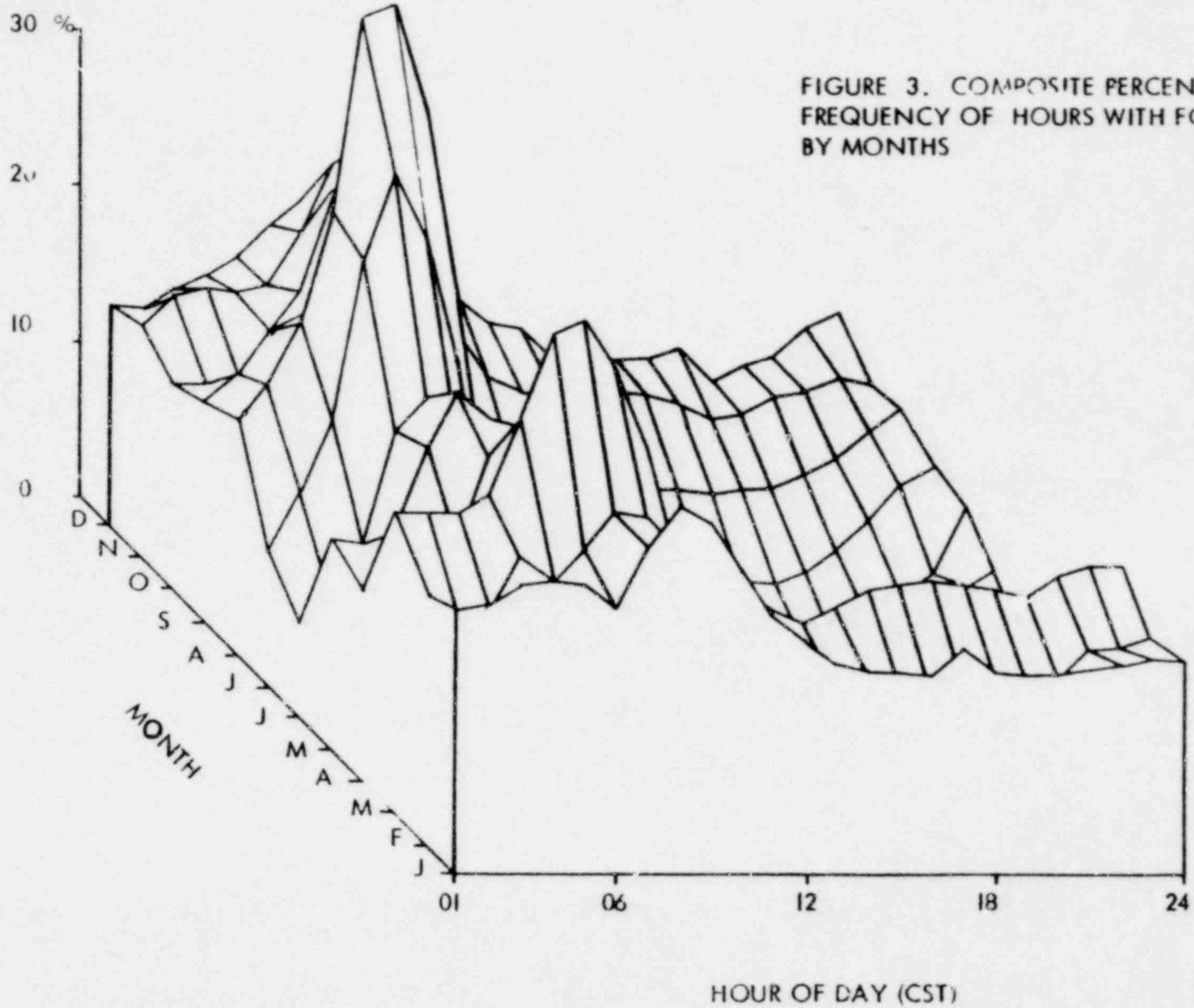


FIGURE 3. COMPOSITE PERCENT FREQUENCY OF HOURS WITH FOG BY MONTHS

TABLE 5

Persistence of "0" Mile Visibility
Joliet Municipal Airport
1958-1970

Persistence (Hours)	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sept.	Oct.	Nov.	Dec.
					(occurrences)							
1	7	2	2	1	1	0	1	5	3	2	4	4
2	4	0	5	1	3	0	1	2	0	1	2	2
3	1	1	4	1	0	0	0	1	0	1	4	3
4	0	0	1	0	0	0	0	1	1	1	1	1
5	1	0	0	1	0	0	0	1	1	0	0	1
6	1	0	0	0	1	0	0	0	0	1	1	0
7	1	0	0	0	0	0	0	1	0	0	0	0
8	0	0	1	0	0	0	0	1	0	0	0	0
9	0	0	0	0	1	0	0	0	0	0	0	0
10	0	0	0	0	0	0	0	0	0	0	0	0
11	0	0	0	0	0	0	0	0	0	0	1	0
12	0	0	0	0	0	0	0	0	0	0	1	0
Maximum Persistence	7	3	8	5	9	0	2	8	5	6	12	5
50% ile	2	1	2	2	2	0	1	2	1	2	3	2
80% ile	3	3	3	5	6	0	2	5	4	4	6	3
90% ile	6	3	4	5	9	0	2	7	5	6	11	4
95% ile	7	3	8	5	9	0	2	8	5	6	12	5
99% ile	7	3	8	5	9	0	2	8	5	6	12	5
99.9% ile	7	3	8	5	9	0	2	8	5	6	12	5

Table Total is 257 observations out a sample of 99,165 hours

IV. Preliminary Study

A pilot project was conducted prior to the preparation of this report. Results as well as techniques used in the anlaysis are summarized here to provide examples of the types of analysis that can be used in the expanded study.

1. Single Station Variability (year to year) in Meteorology as it Affects Dispersion Estimates.

1.1 ANL meteorological data were used to generate stability wind roses for each year of the 5-year period 1965-1969. Wind velocity was measured at 150 ft. above grade; stability was based on differential temperatures measured at 144 ft. and 5 ft. above grade.

The data were input to Meteorology and Atomic Energy equation 7.63 * along with the appropriate physical and operating characteristics of the Dresden Nuclear Power Station (DNPS). Annual gamma dose values were then calculated for two ground-level locations, one with a bearing of 30 degrees and the other with a bearing of 210 degrees. **

* Meteorology and Atomic Energy-1968. USAEC Division of Technical Information.

** This study was reported in CECO Dresden Report No. 21, Measurements of Radioactivity in Process Systems of the DNPS and in the Environment, January-February 1971.

The meteorological variations from year to year resulted in gamma dose levels varying by a factor of 1.22 at the 30-degree bearing location, and 1.88 at the 210-degree bearing location.

Evaluations of ground-level dose were not made in other directions, or locations.

1.2 Maximum off-site ground-level relative concentration values (X/Q) were calculated for the DNPS. ANL data were used to generate stability wind roses for each of the years 1965 through 1969. Wind velocity measurements at 150 ft. were used. Stability was based on differential temperatures measured at 144 ft. and 34 ft. above grade. The dispersion equation was evaluated at several downwind distances. The variation in the maximum annual average values was about twenty percent. On the other hand, the variation between a given maximum average annual value and values in the other years at the same geographical point as the maximum was at least 1.5.

The results of the studies described above suggest that year to year meteorological variations at a given site may result in variations in calculated ground-level gamma doses and/or concentrations amounting to a factor of two.

2. Geographical Variation in Meteorology as it Affects Dispersion Estimates

2.1 Stability wind rose data from QUAD and DNPS representing the period January-June 1973 were factored into the dispersion equation to determine the variation in "X/Q". The maximum value based on the DNPS wind rose was 1.9×10^{-8} Sec/m^3 whereas the corresponding value based on the QUAD wind rose was 2.6×10^{-8} Sec/m^3. The ratio of the QUAD to DNPS was, thus 1.37.

The results of this study tend to suggest that the maximum annual average off-site relative concentration factors or doses at two different sites differ from one another by amounts less than or equal to the variations experienced at a single site over a period of several years.

3. Short Term Comparability of Meteorological Observations Between Sites

3.1 Time Series

The first step taken in the study was to visually display the meteorological observations to grasp qualitatively the relationships that might exist between the measurements. For this purpose five days were chosen arbitrarily in the summer, namely 4 June 1973 through 8 June 1973. The data, already stored on magnetic tape, were input to a Calcomp Plotter program which generated figures 4 through 13 .

The plots revealed some patterns and similarities among certain of the measurements at each station as well as between stations.

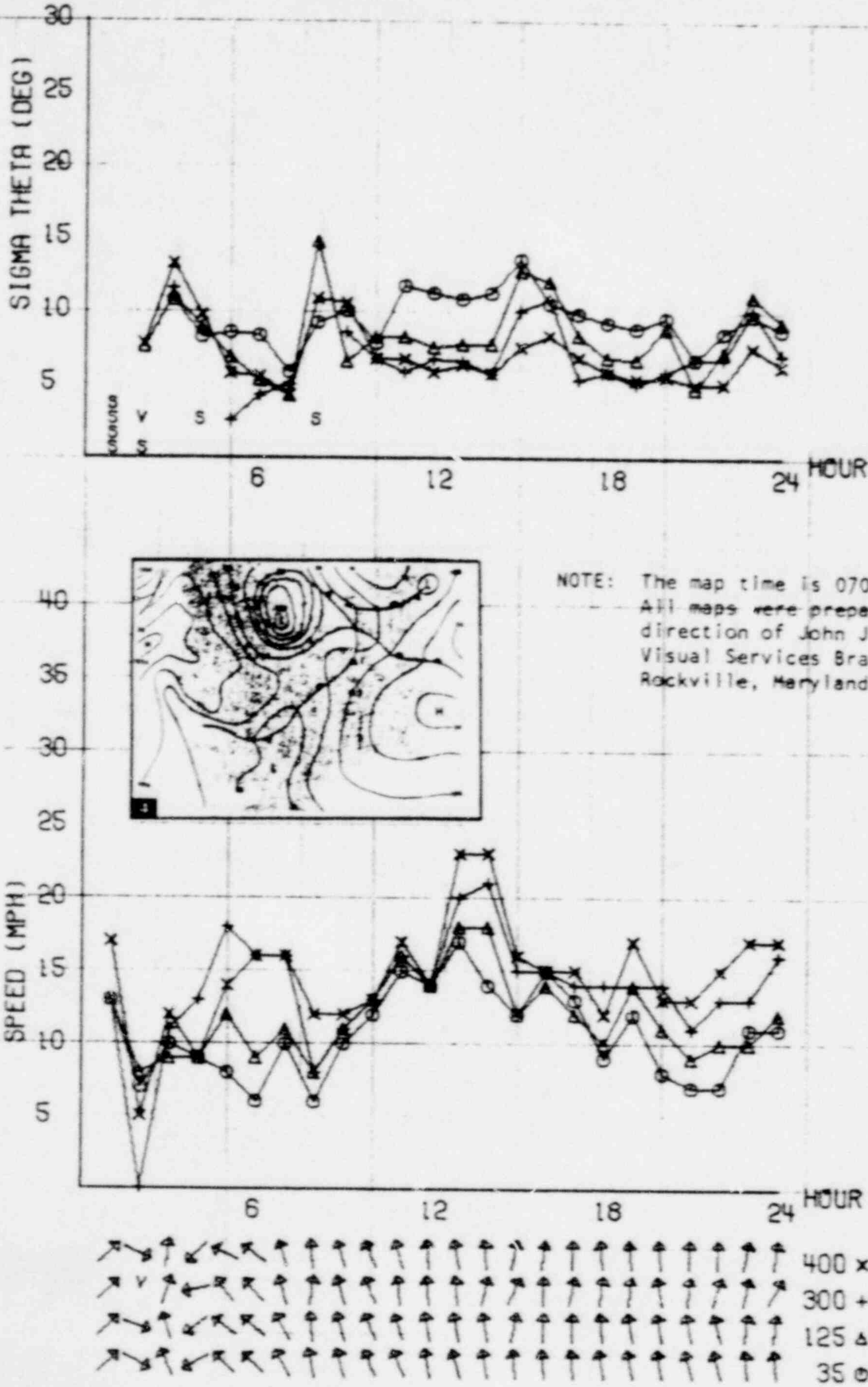
For example, wind direction, especially at the upper elevations appeared to be quite uniform for long periods of time interrupted for relatively short periods by apparently systematic changes. These features were evident at both sites. Usually wind shifts at DNPS lagged behind those at QUAD by one or two hours. Frontal passages accounted for the majority of the major wind shifts as indicated by the daily weather map. Causes for all the direction shifts were not pursued but it was felt that the major change in direction could be adequately explained either in terms of frontal passages, squall lines, thunderstorms or the like. The significant feature at this point of the study was the "relatedness" of the wind direction at the two sites.

Also evident in the time-series was the apparently strong relatedness between wind direction at different elevations. Typically the wind veered with height. This was more noticeable between the 35 ft. wind and the rest than between the upper winds. Apparently the 35 ft. elevation was substantially affected by the ground and other frictional influences whereas the other elevations were not. Further indications of the 35 ft. wind responses to trees and other ground-effects were found in the sigma-theta values which increased considerably with a south to west wind at DNPS, the effect, no doubt, of the tree stands upwind of the sensor in those directions.

The wind speed variations with height also appeared to be related well, in this case to atmospheric stability as evidenced by the diurnal variations at all the elevations. Wind shear was often strong at night and weak during the day. This relationship was partially masked at times and completely masked at others by the large-scale synoptic influences depicted in the weather maps.

Patterns were then sought in winter data. For this purpose the data were plotted for 15 January 1973 through 19 January 1973. These plots are shown in figures 14 through 23 . Once again some patterns were evident.

Clearly, the time-series indicated that the search for, and application of, quantitative relationships should prove meaningful and worthwhile, not only between the measurements at one site but also between those from the two sites.



NOTE: The map time is 0700 EST.
All maps were prepared under the direction of John J. Smiles, Chief, Visual Services Branch, NOAA, Rockville, Maryland, for Weatherwise.

Figure 4. Daily Weather Summary

QUAD 6/ 4/ 73

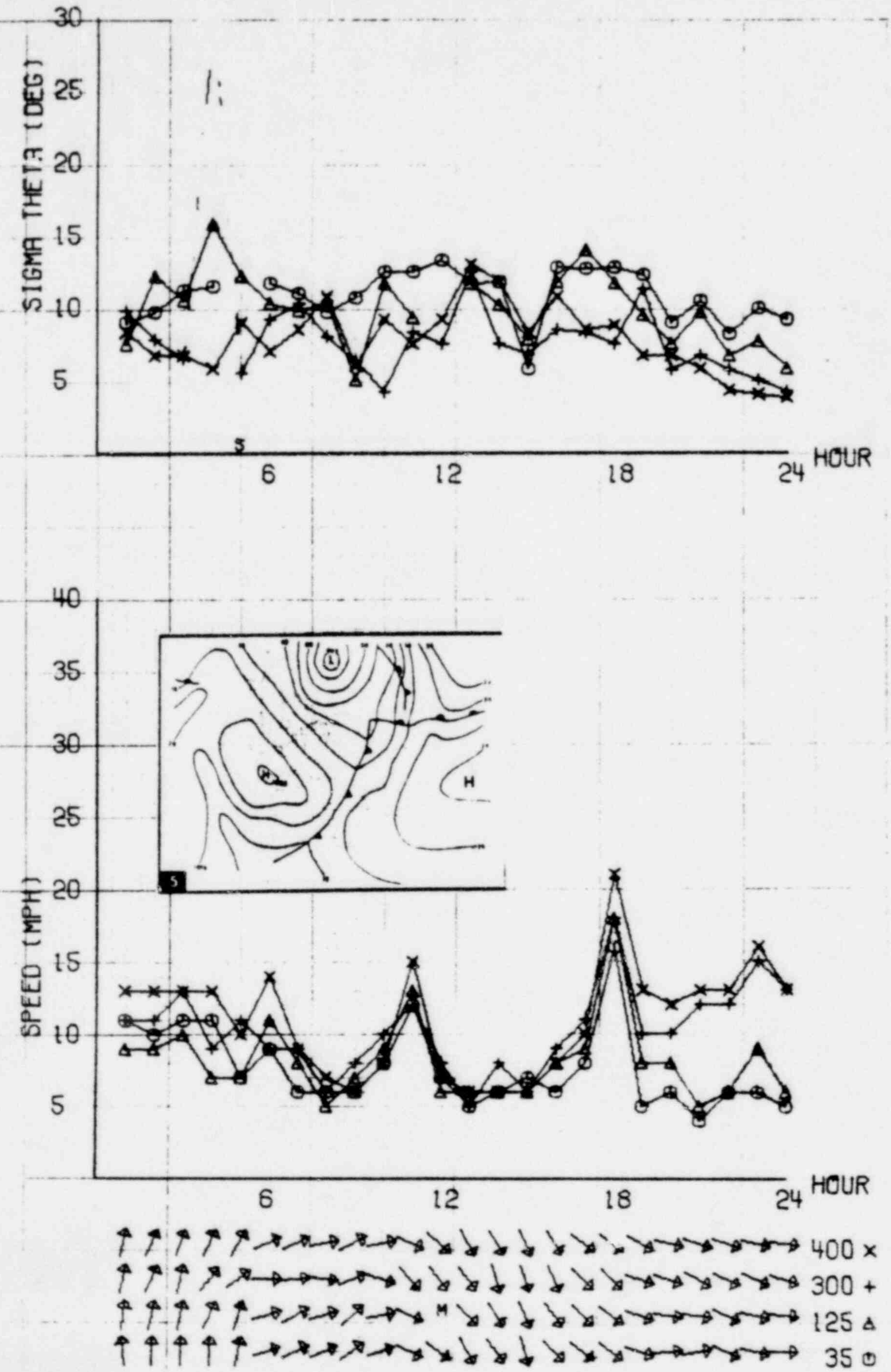


Figure 5. Daily Weather Summary
QUAD 6/ 5/ 73

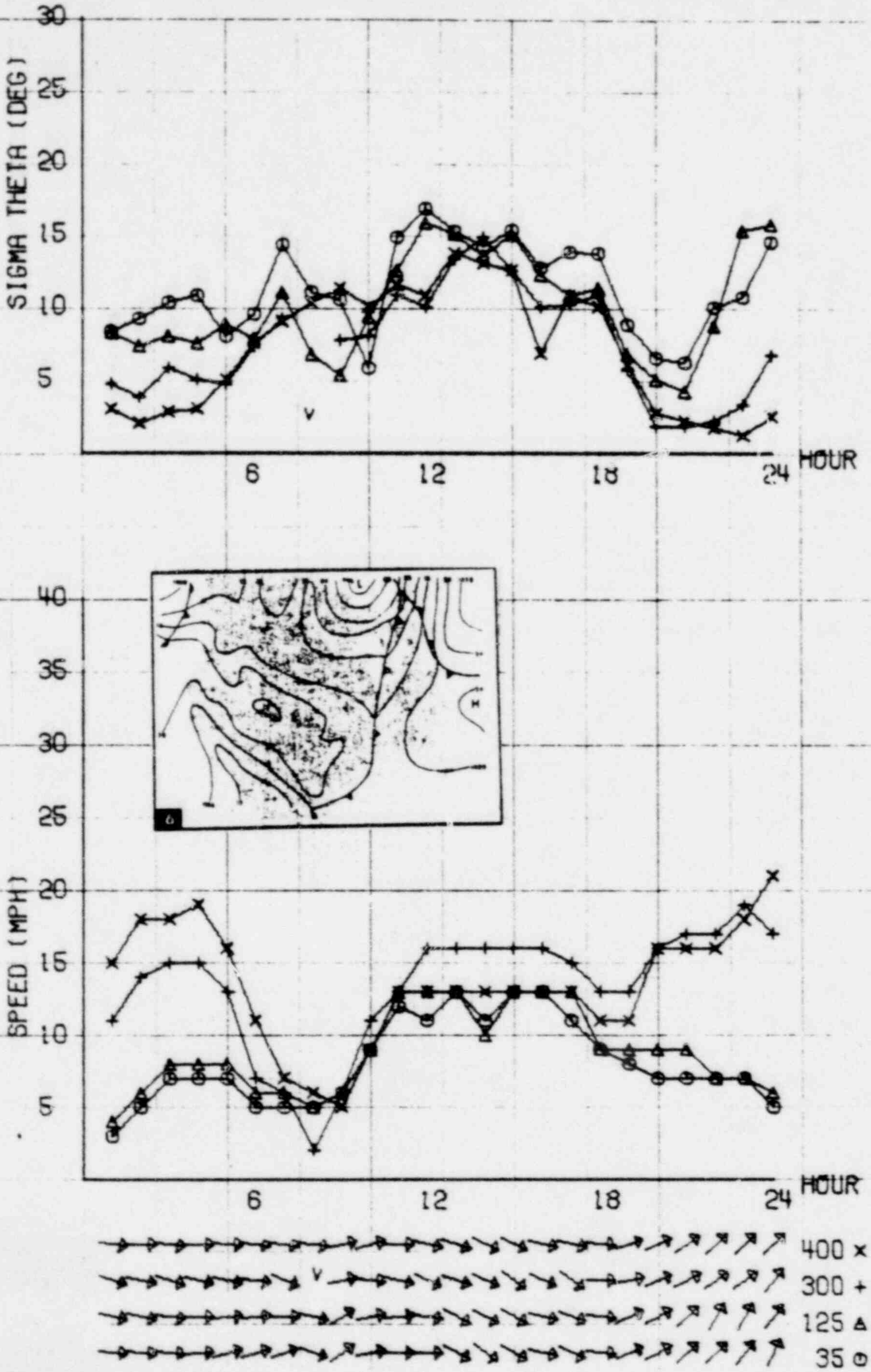


Figure 6. Daily Weather Summary

QUAD 6/ 6/ 73

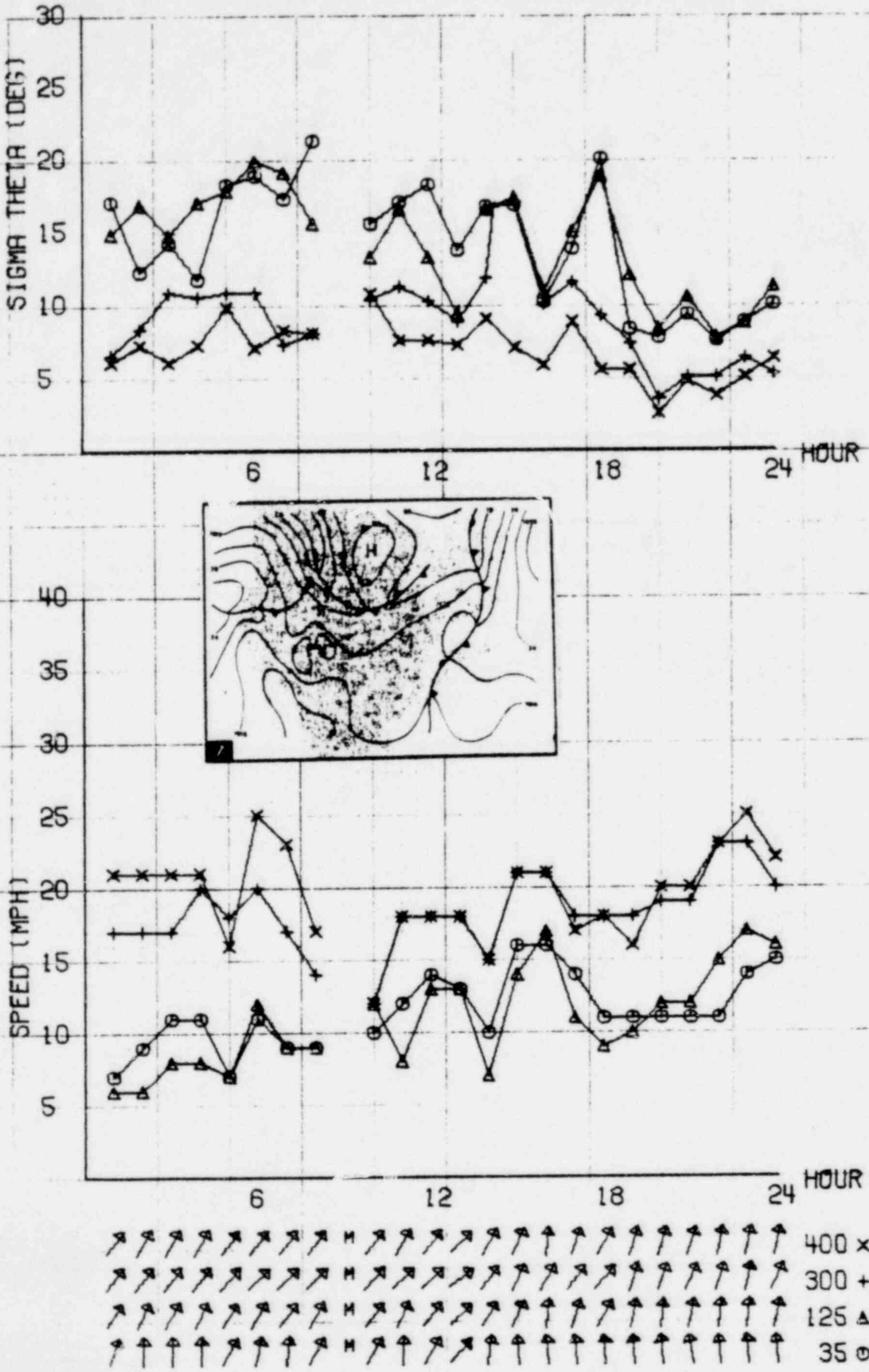


Figure 7. Daily Weather Summary

QUAD 6/ 7/ 73

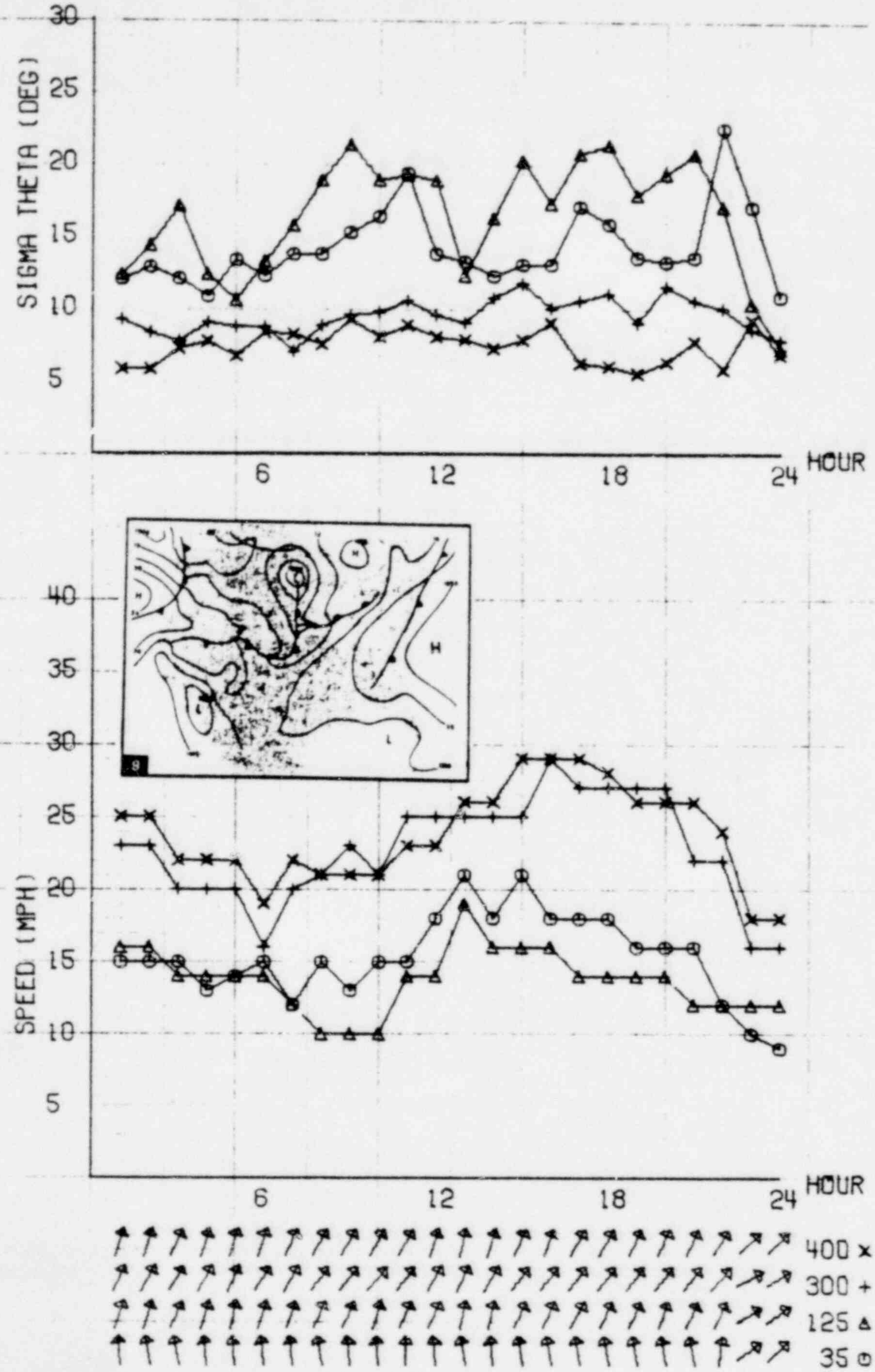


Figure 8. Daily Weather Summary

QUAD 6/ 8/ 73

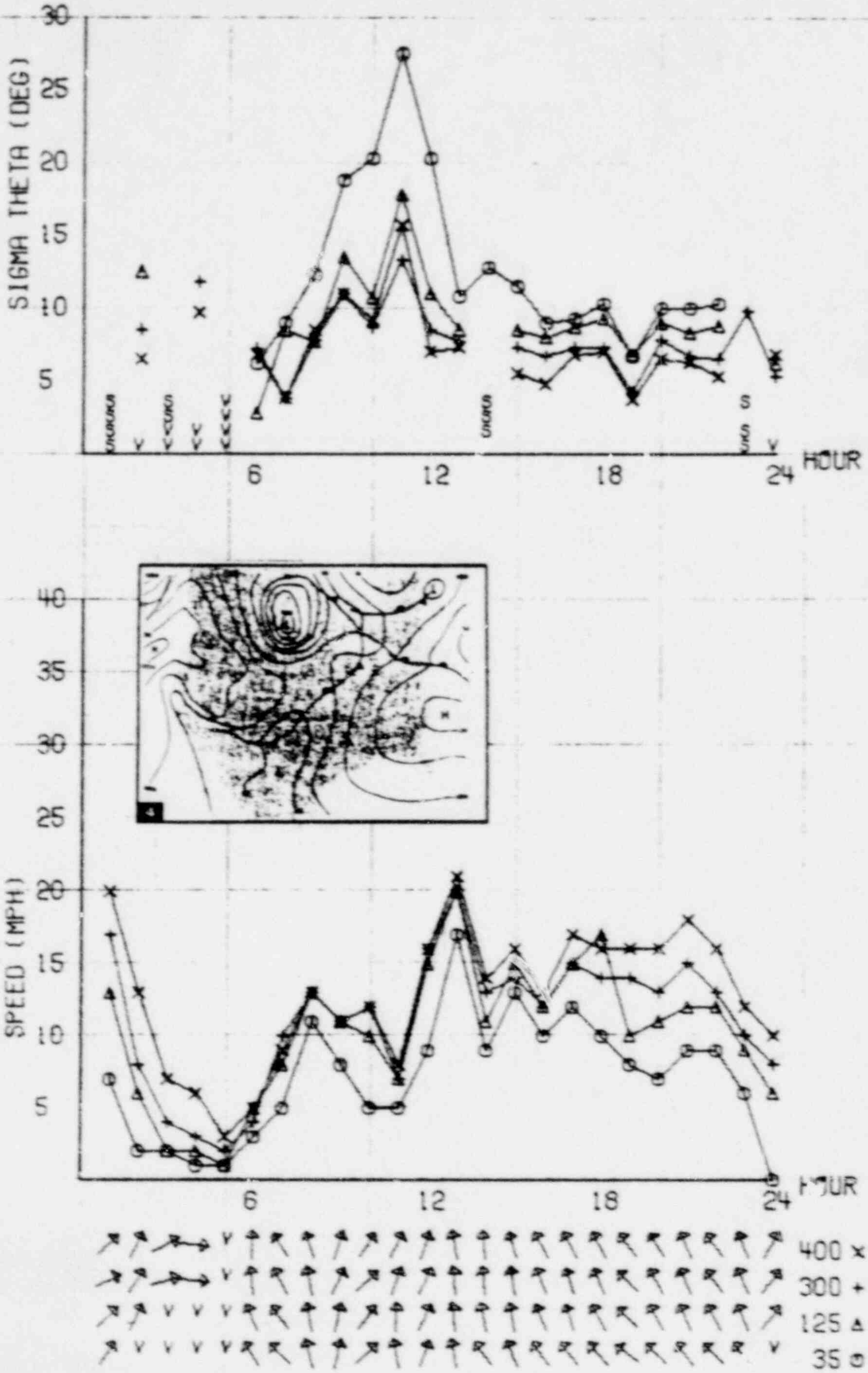


Figure 9. Daily Weather Summary

DNPS 6/ 4/ 73

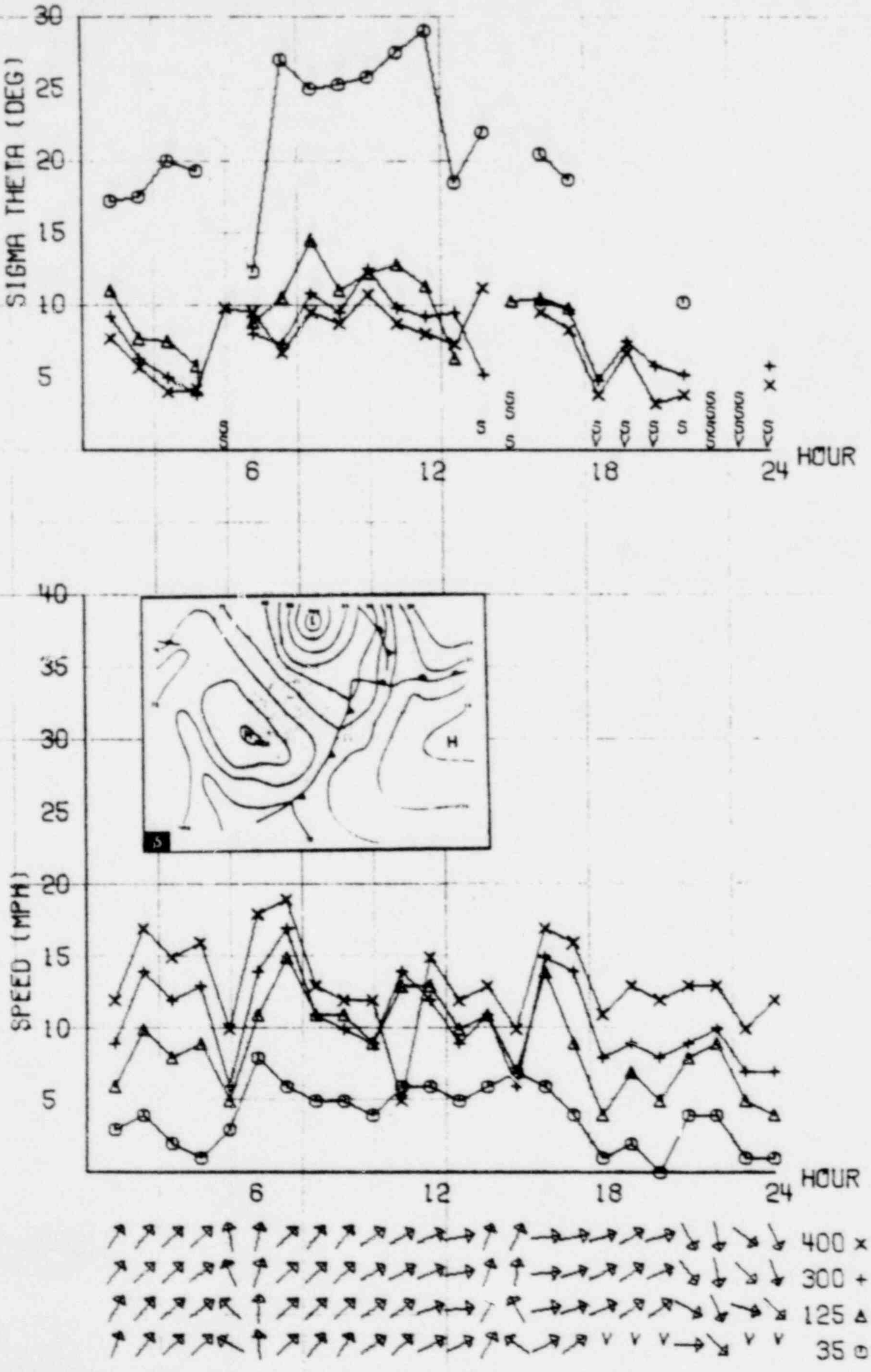


Figure 10. Daily Weather Summary

DNPS 6/ 5/ 73

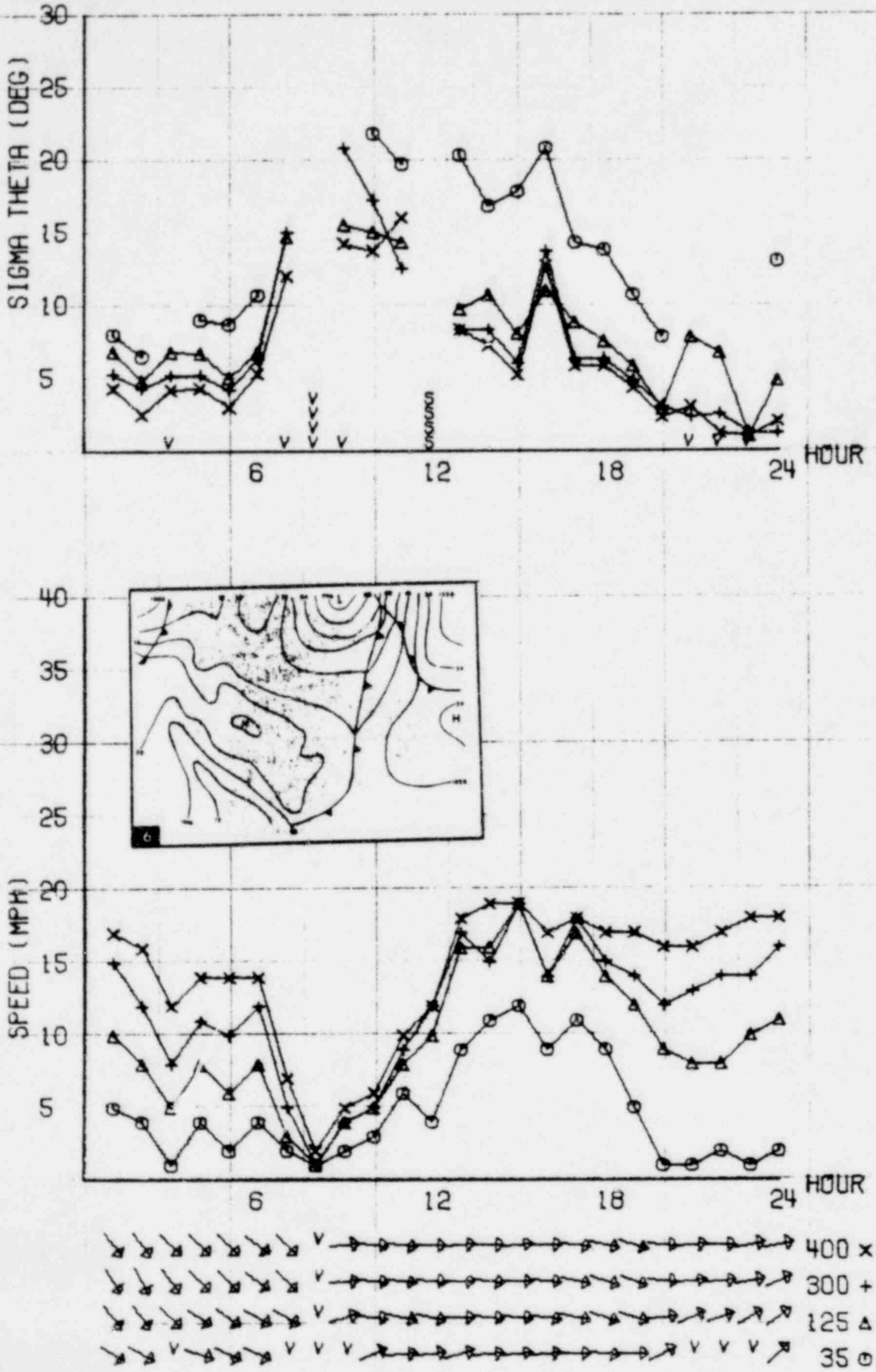


Figure 11. Daily Weather Summary

DNPS 6/ 6/ 73

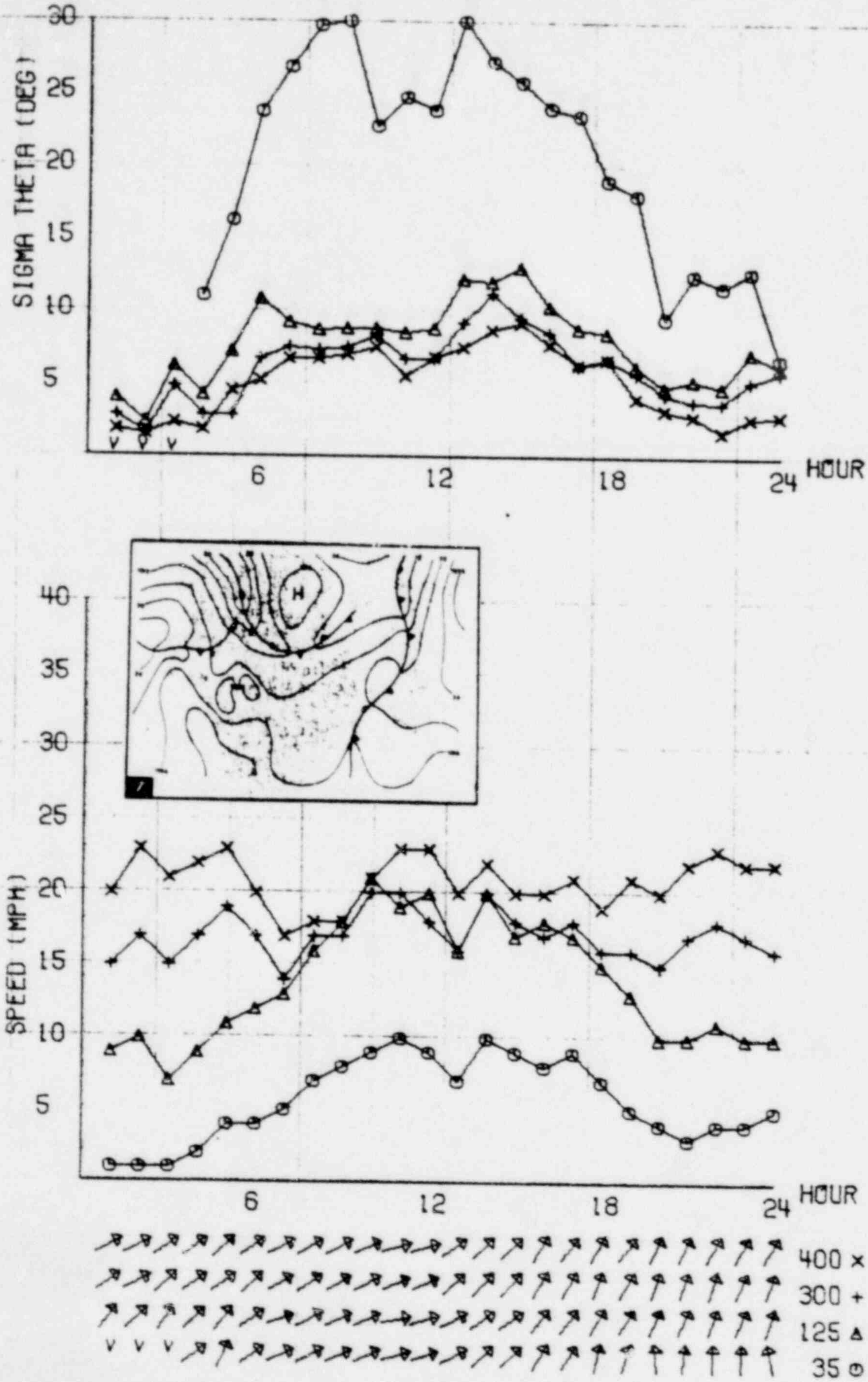


Figure 12. Daily Weather Summary

DNPS 6/ 7/ 73

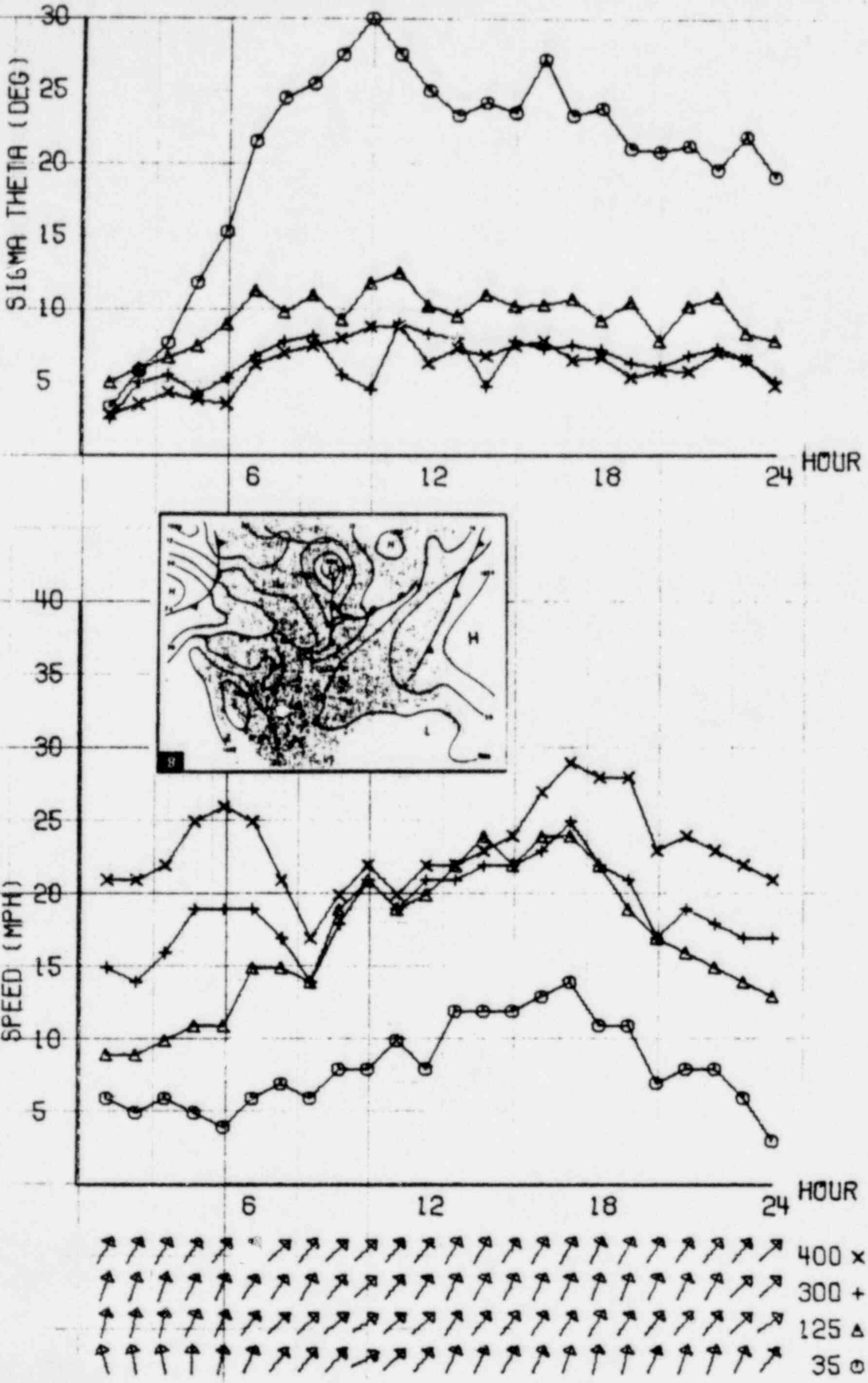
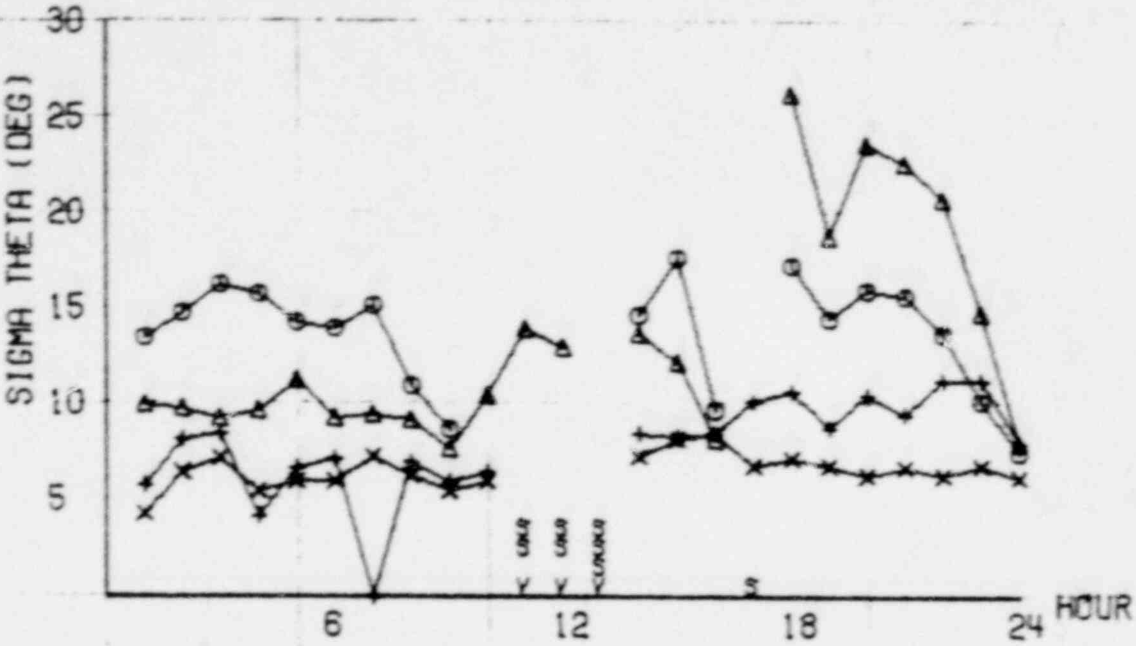


Figure 13. Daily Weather Summary

DNPS 6/ 8/ 73



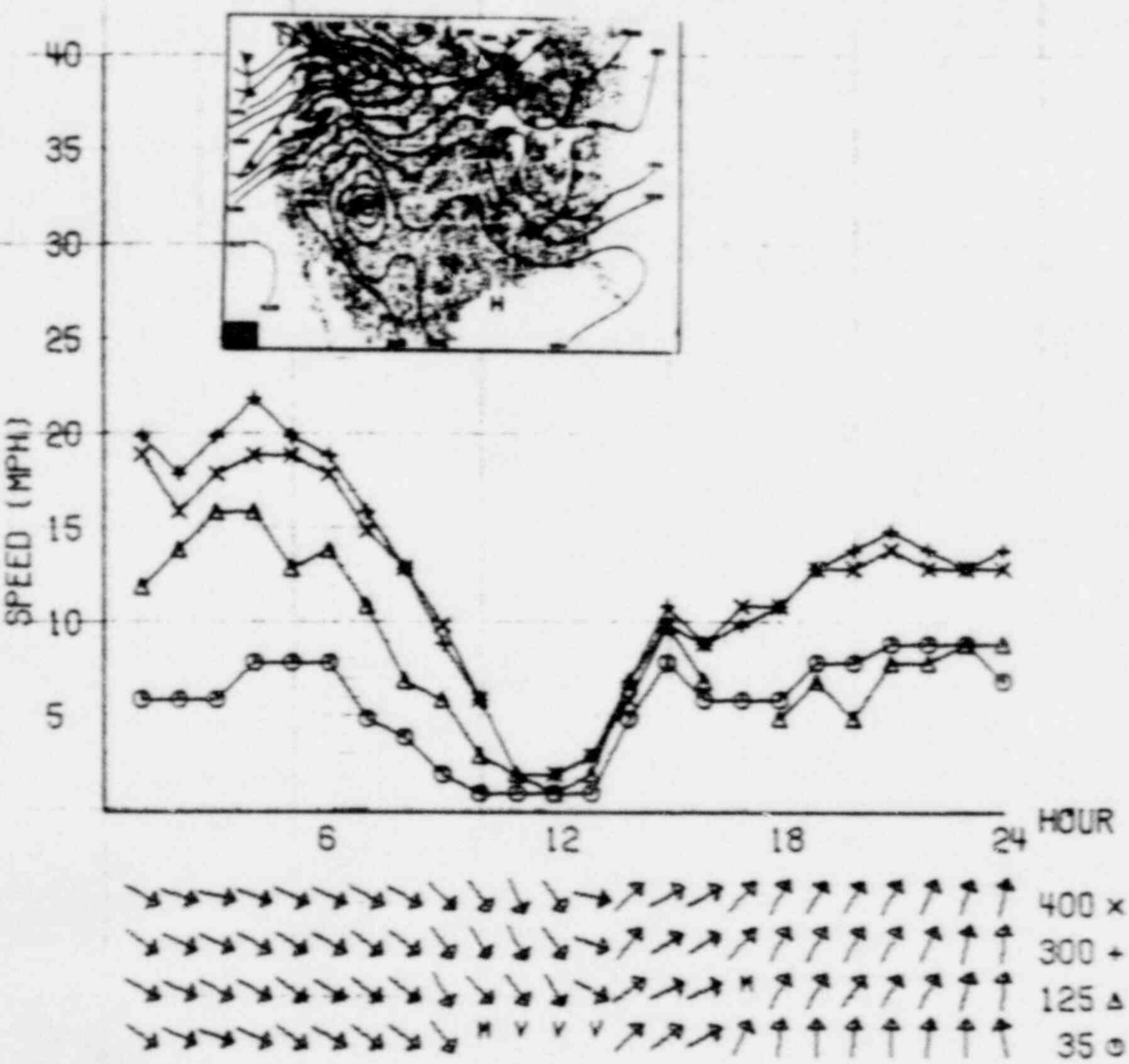


Figure 14. Daily Weather Summary

QUAD 1/ 15/73

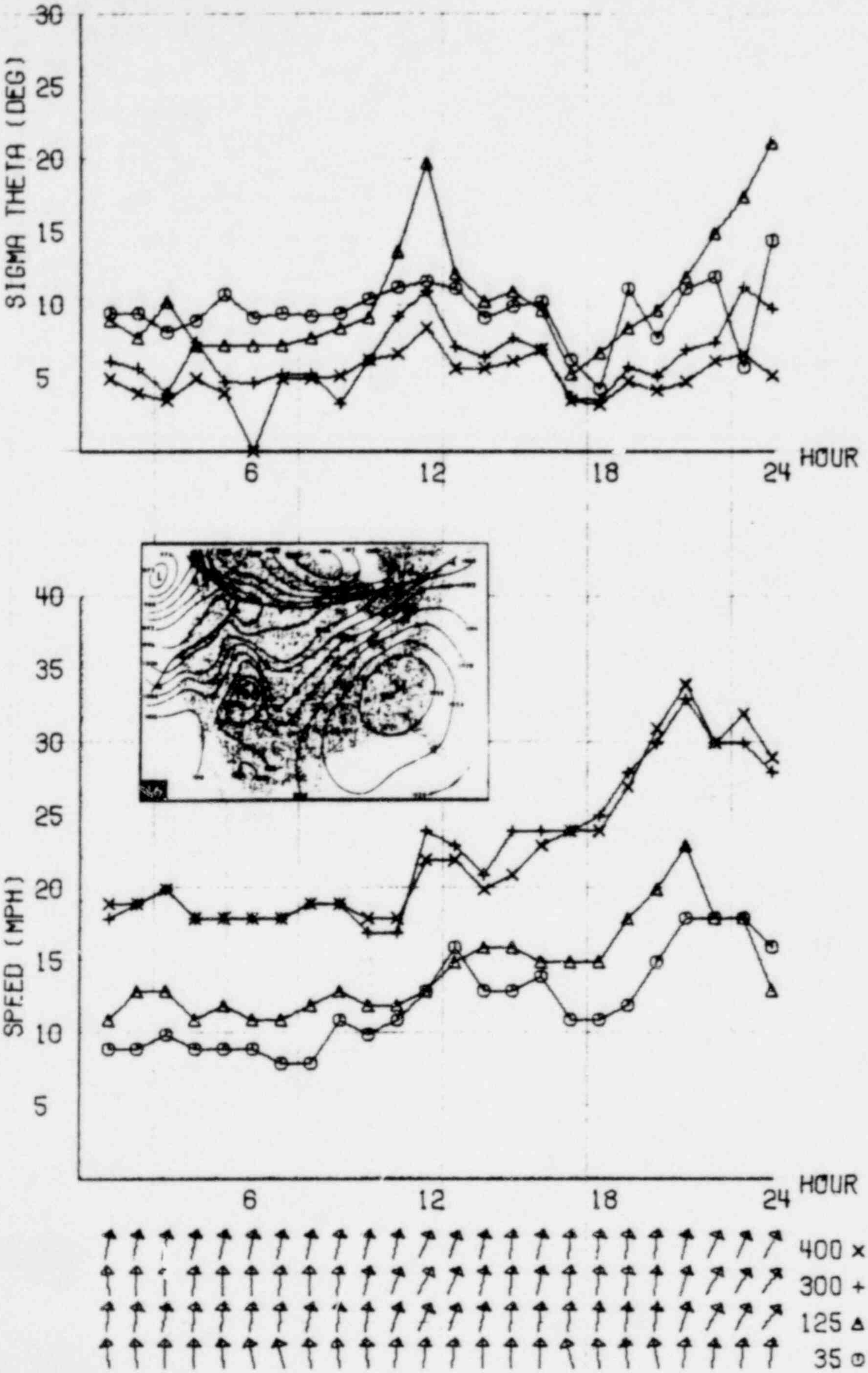


Figure 15. Daily Weather Summary

QUAD 1/ 16/73

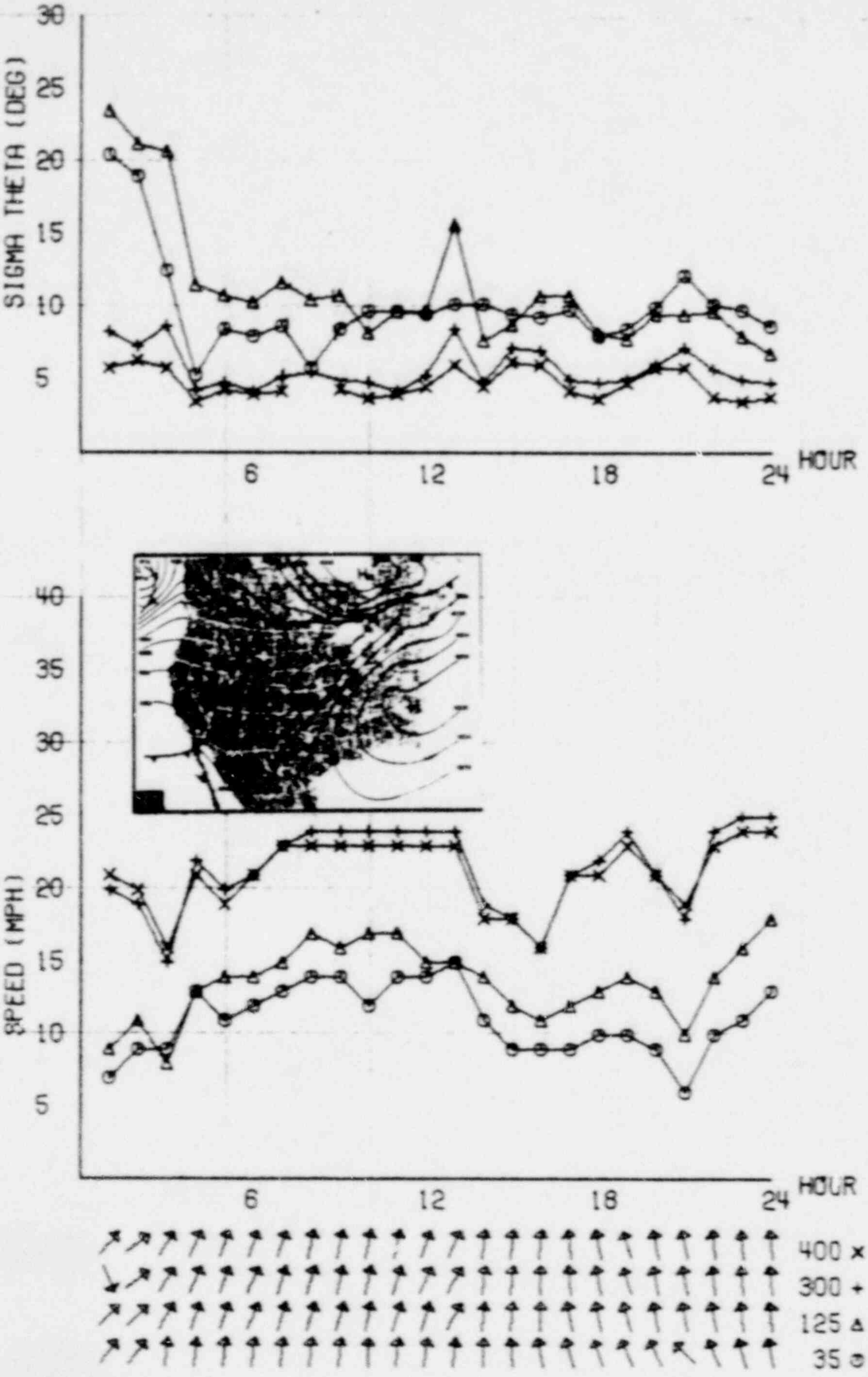


Figure 16. Daily Weather Summary

QUAD 1/ 17/73

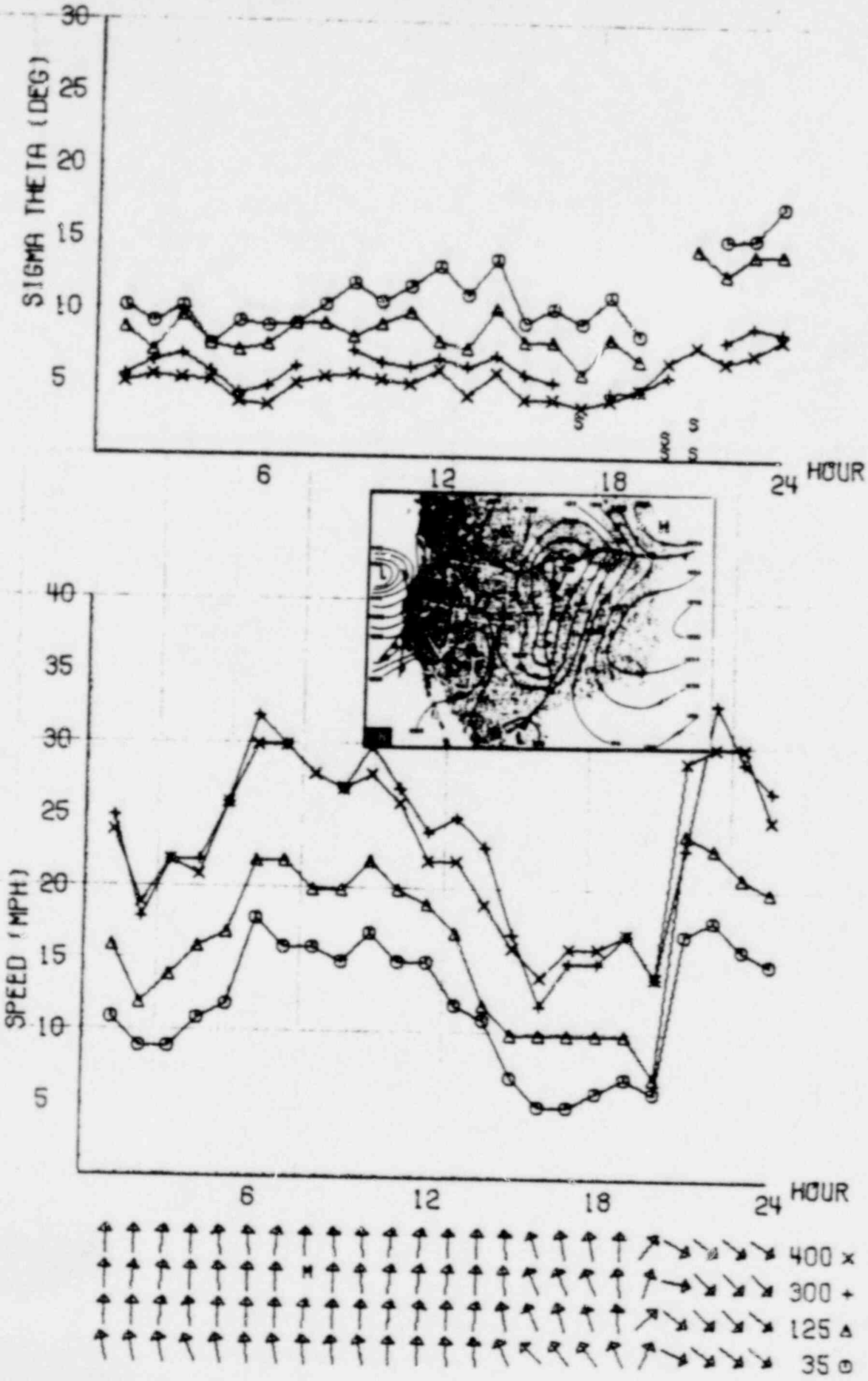
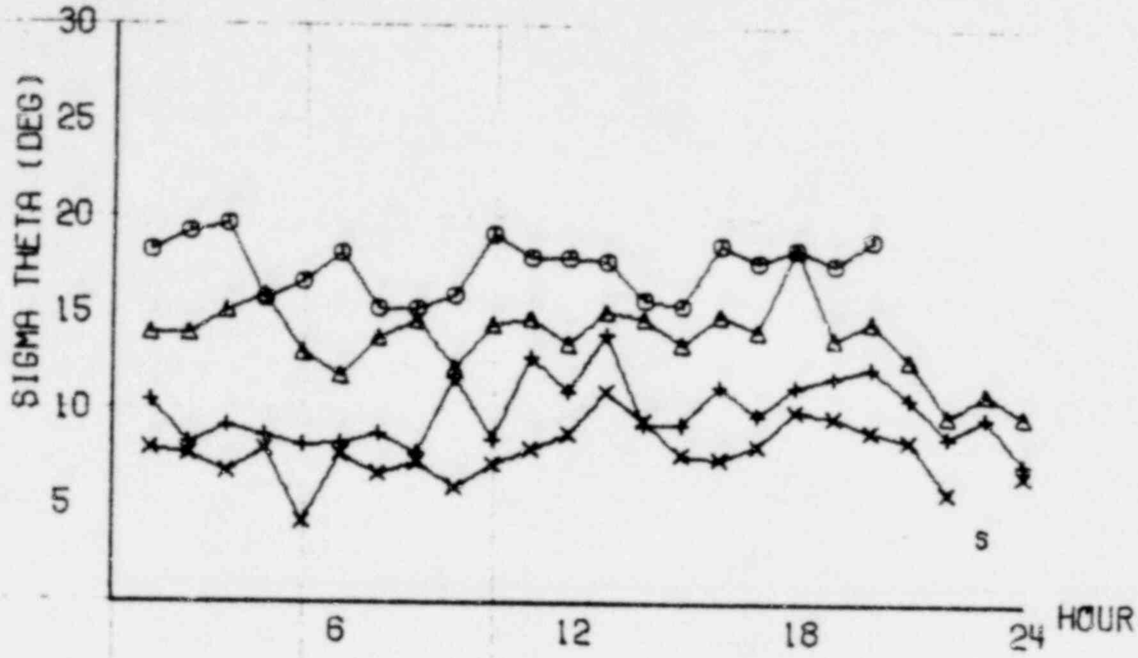


Figure 17. Daily Weather Summary

QUAD 1/ 18/73



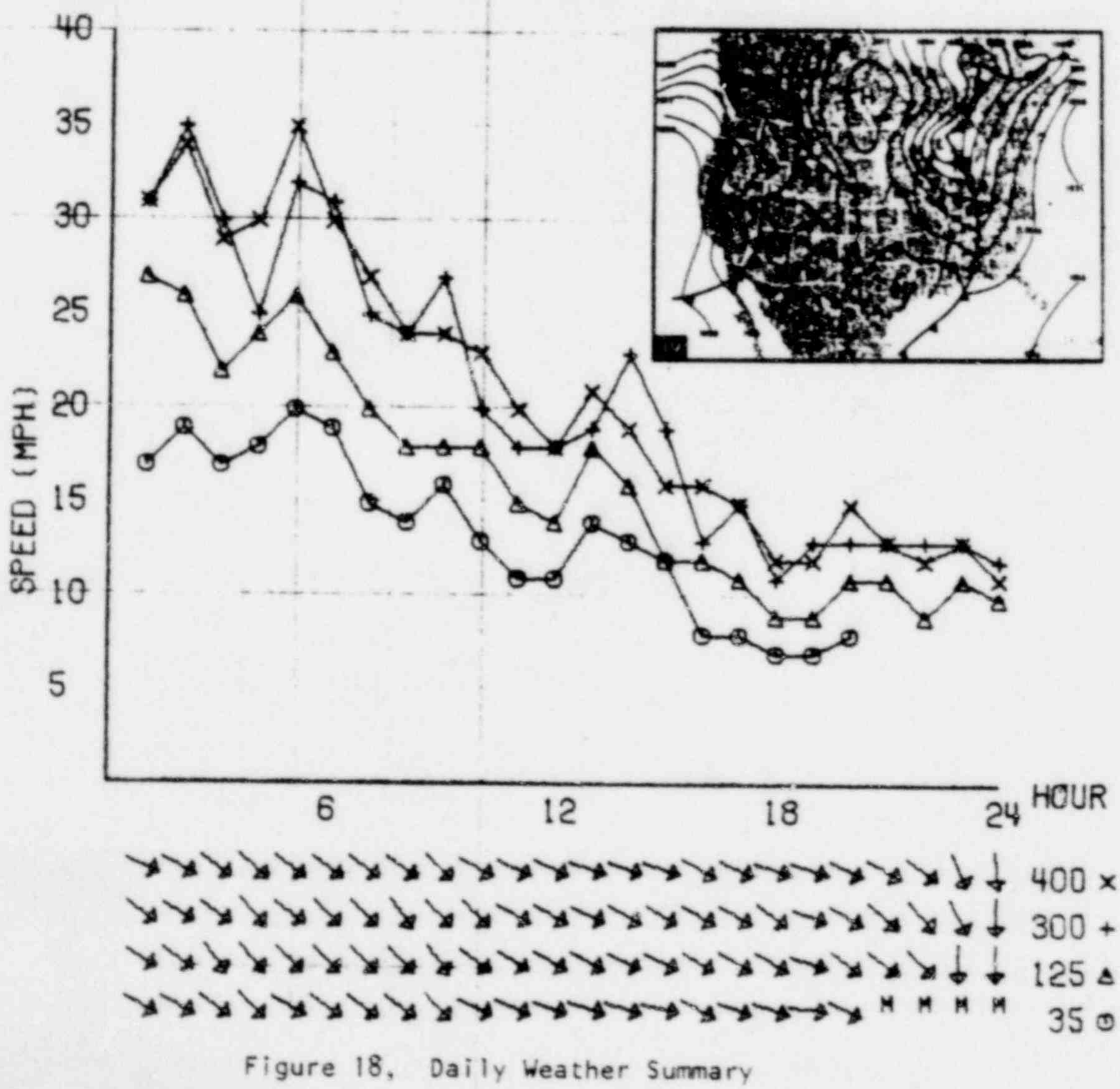


Figure 18, Daily Weather Summary

QUAD 1/ 19/73

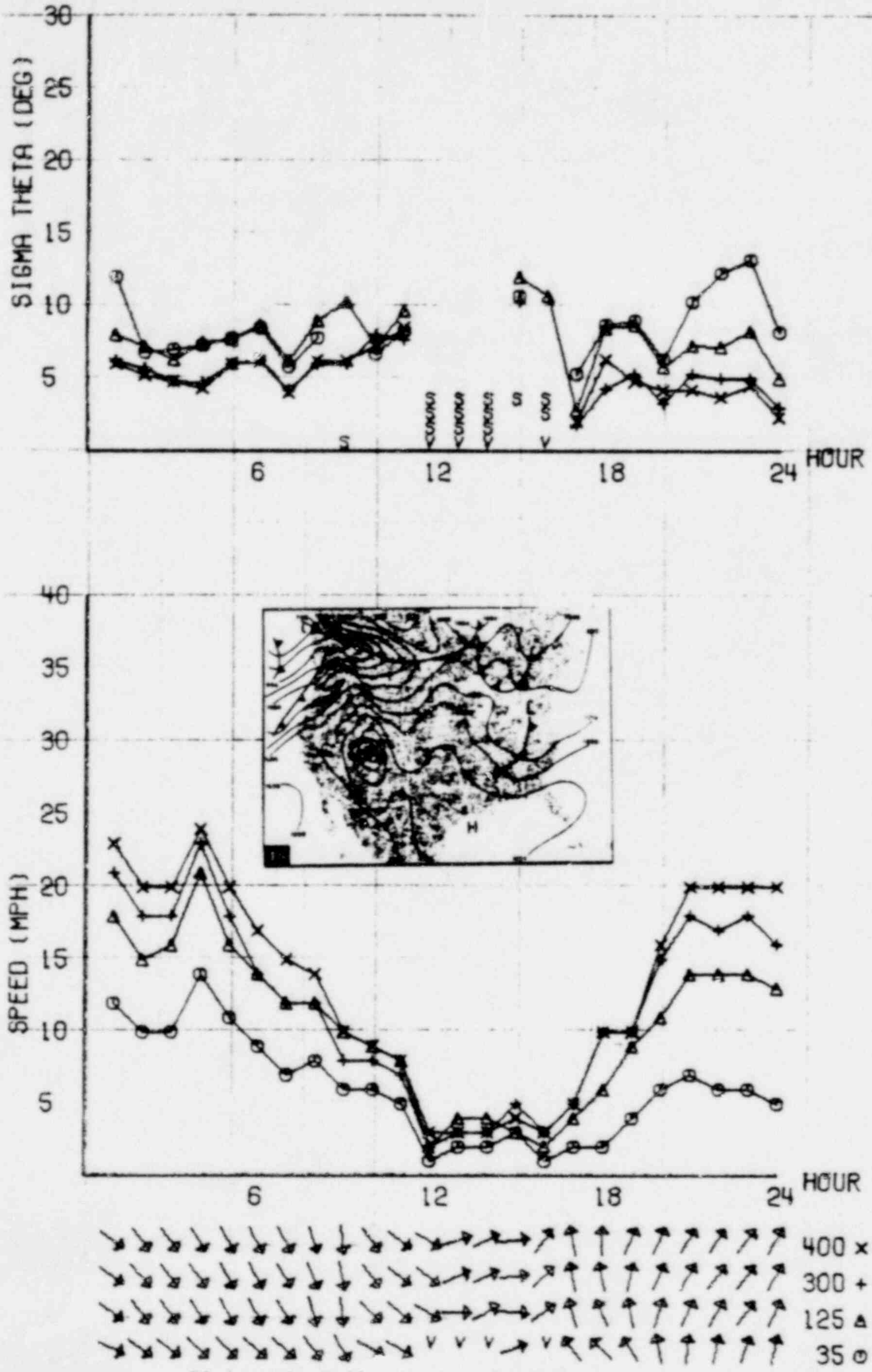


Figure 19, Daily Weather Summary

DNPS 1/ 15/73

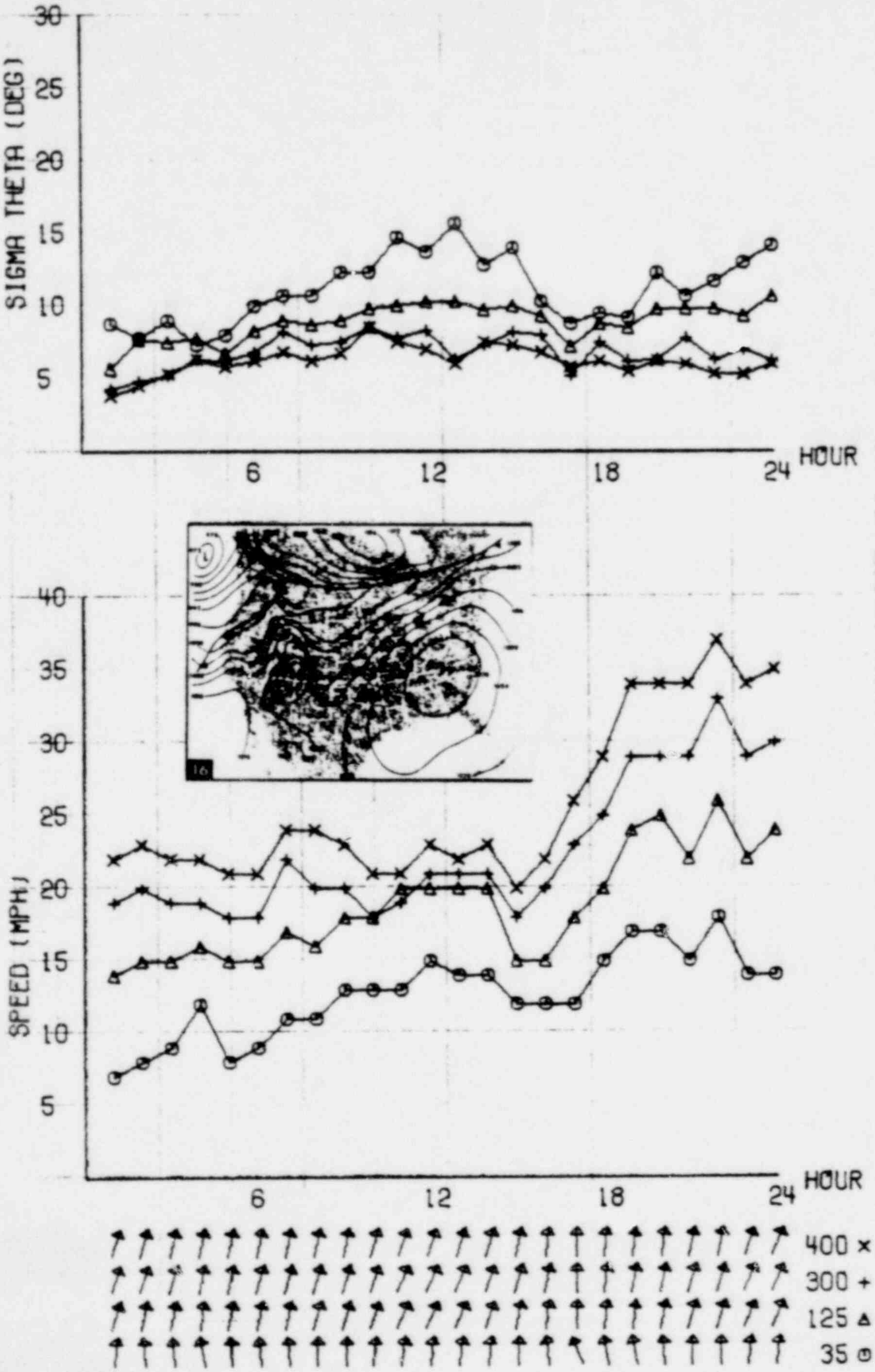


Figure 20. Daily Weather Summary

DNPS 1/ 16/73

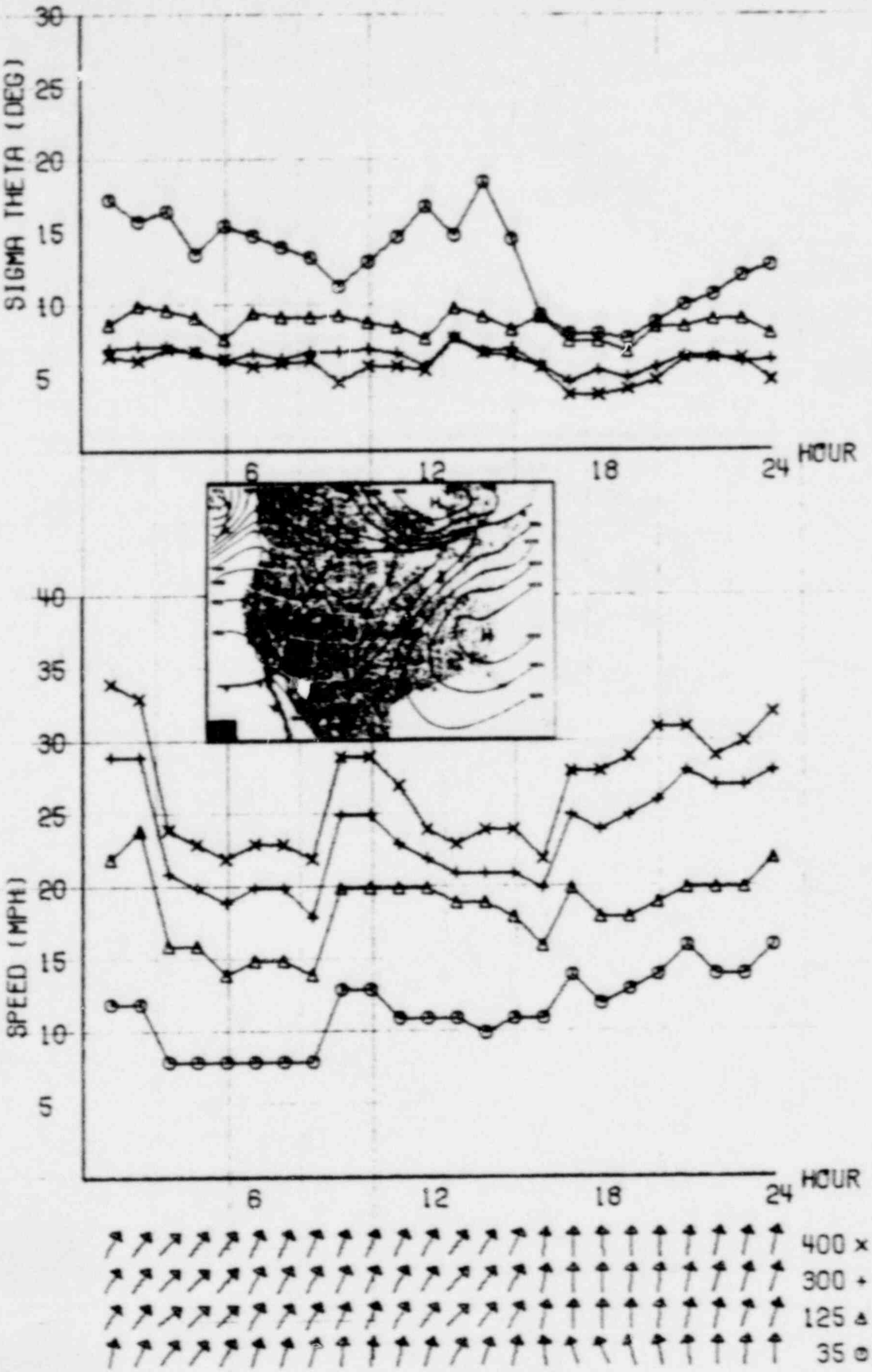


Figure 21. Daily Weather Summary

DNPS 1/ 17/73

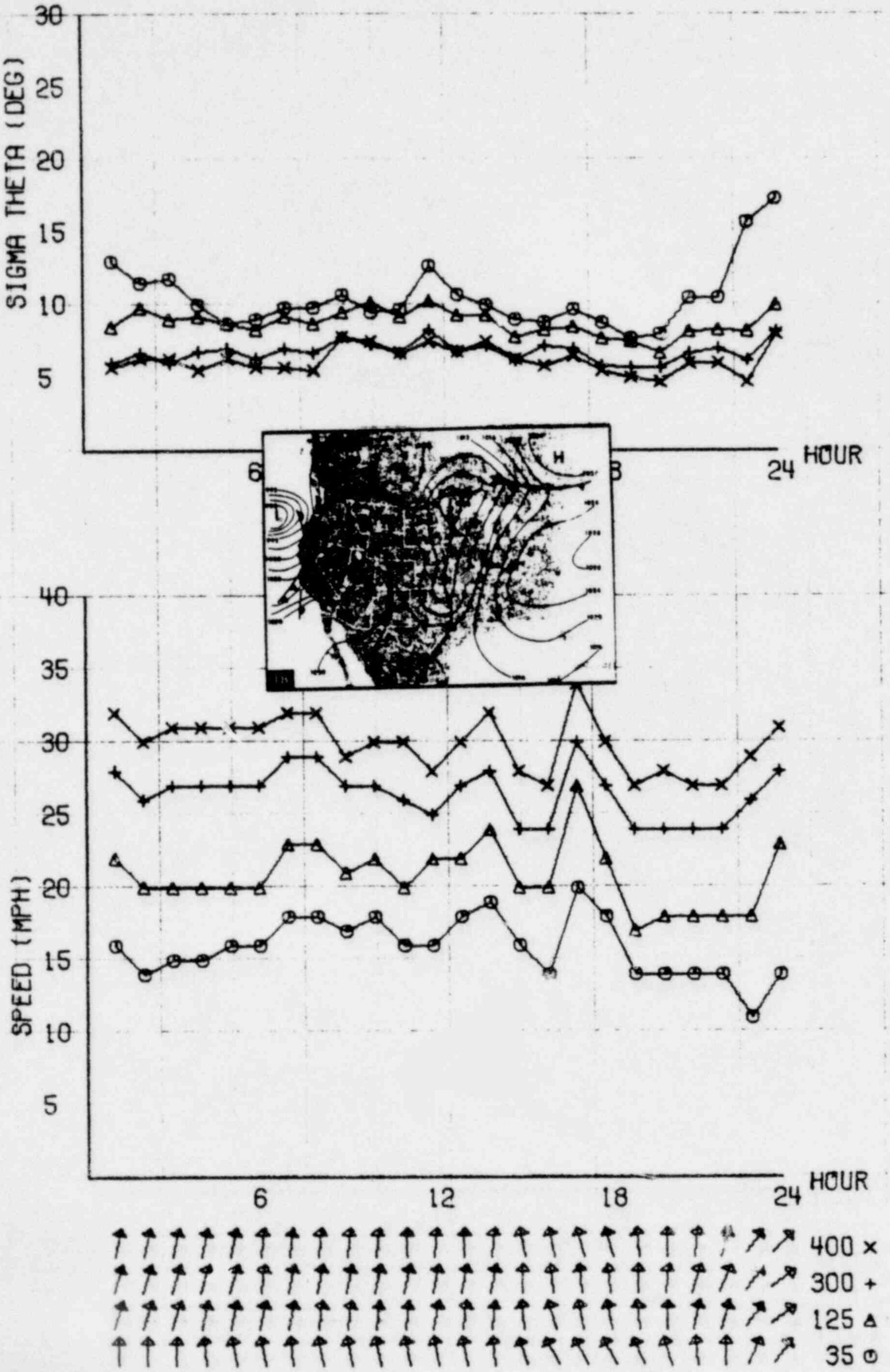


Figure 22. Daily Weather Summary

DNPS 1/ 18/73

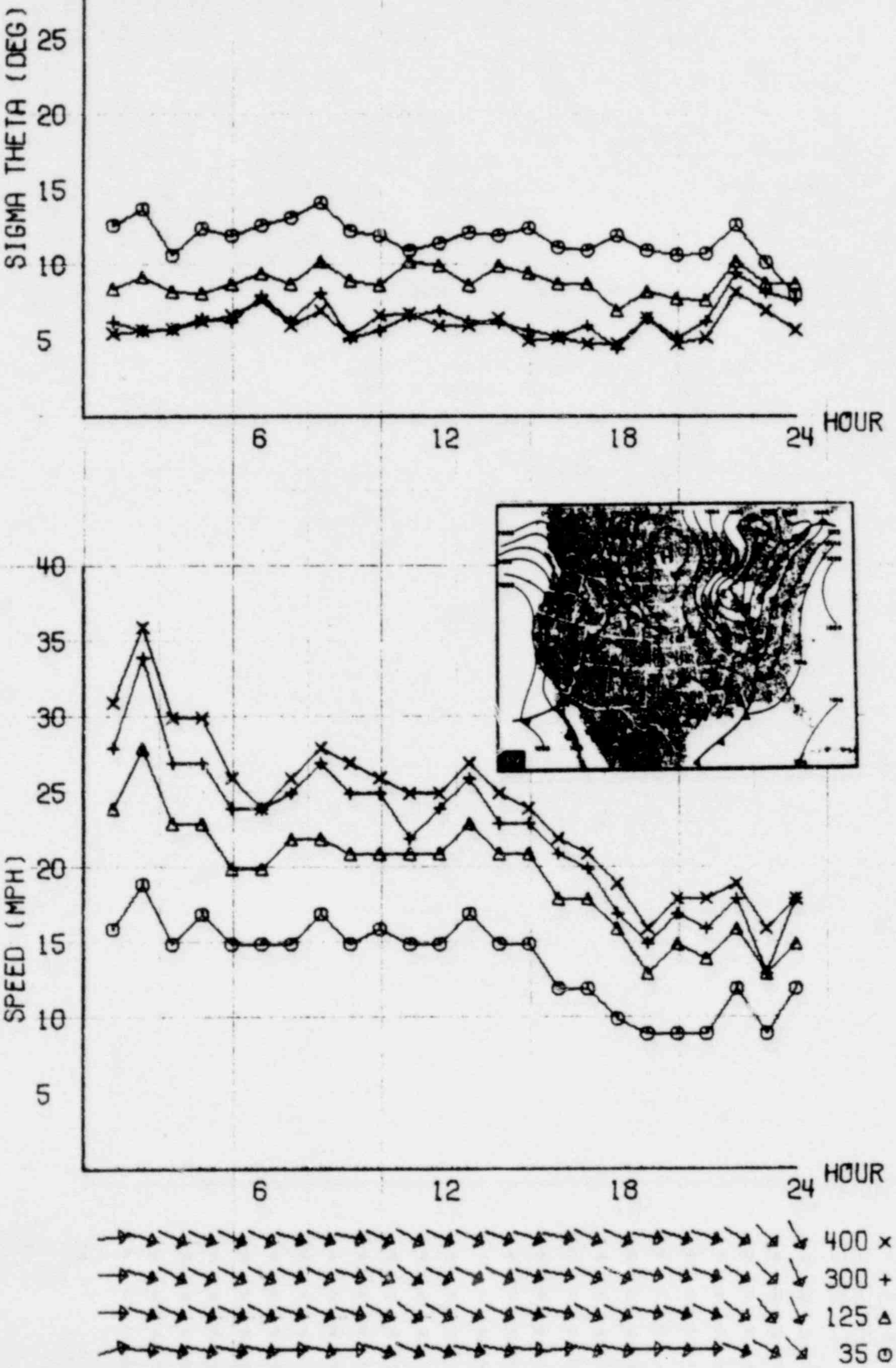


Figure 23. Daily Weather Summary

DNPS 1/ 19/73

3.2 Quantitative Relationships Between Measurements at Two Sites

Preliminary attempts were made to derive objective methods to describe the similarities made intuitively clear by the time series plots. They are described in the following.

3.2.1 400 ft. elevation wind direction

The hourly wind direction values commencing at 0100 CST on 4 June 1973 and ending on 2400 CST on 7 June 1973 at DNPS and QUAD was compared by generating a variety of statistics given in Table 6. The average direction at the two sites differed by only 5 degrees.

TABLE 6

Comparative Statistics for DNPS and QUAD
4 June through 7 June 1973
400 ft. Wind Direction (degrees)

Name	Value	
	DNPS	QUAD
Mean	234.4	229.1
Standard Deviation	48.7	53.3
Maximum	349	339
Minimum	143	44
Range	206	295
Skewness [1]	0.121	-0.230
Kurtosis [2]	-0.364	0.260

(footnotes on following page)

1. A distribution is "skewed" when there is a considerably larger number of extreme cases on one side of the distribution curve than on the other. A positive value indicates extremely large values are farther from the mean than extremely small values are.

2. "Kurtosis" is a measure of the general "peakedness" of the distribution. A positive value indicates the distribution is more peaked than the "normal" distribution.

The averaged data compared quite well. Comparison of individual hourly measurements would not be expected to show as high a measure of similarity, the major reasons being the presence of fronts, squall lines, thunderstorms or the like that affected an observation at one of the sites but not at the other. Never-the-less a linear regression made of the DNPS hourly directions on the QUAD hourly directions yielded a correlation coefficient of 0.58. Greater than 34 percent of the variance in the data was "explained" by the regression equation (figure 24). Removal of those observations affected by fronts, etc. would certainly improve the correlation. For example, the removal of three observations affected by thunderstorms improved the correlation coefficient to 0.74 and 55 percent of the variance was "explained" by the regression equation. Moreover, the standard error of estimate improved from 40 to 33 degrees. Removal of other anomalies would no doubt improve the correlation even more.

(DOWN) DDIR400 DNPS 400 FT WIND DIRECTION JUNE
(ACROSS) QDIR400 QUAD 400 FT WIND DIRECTION JUNE

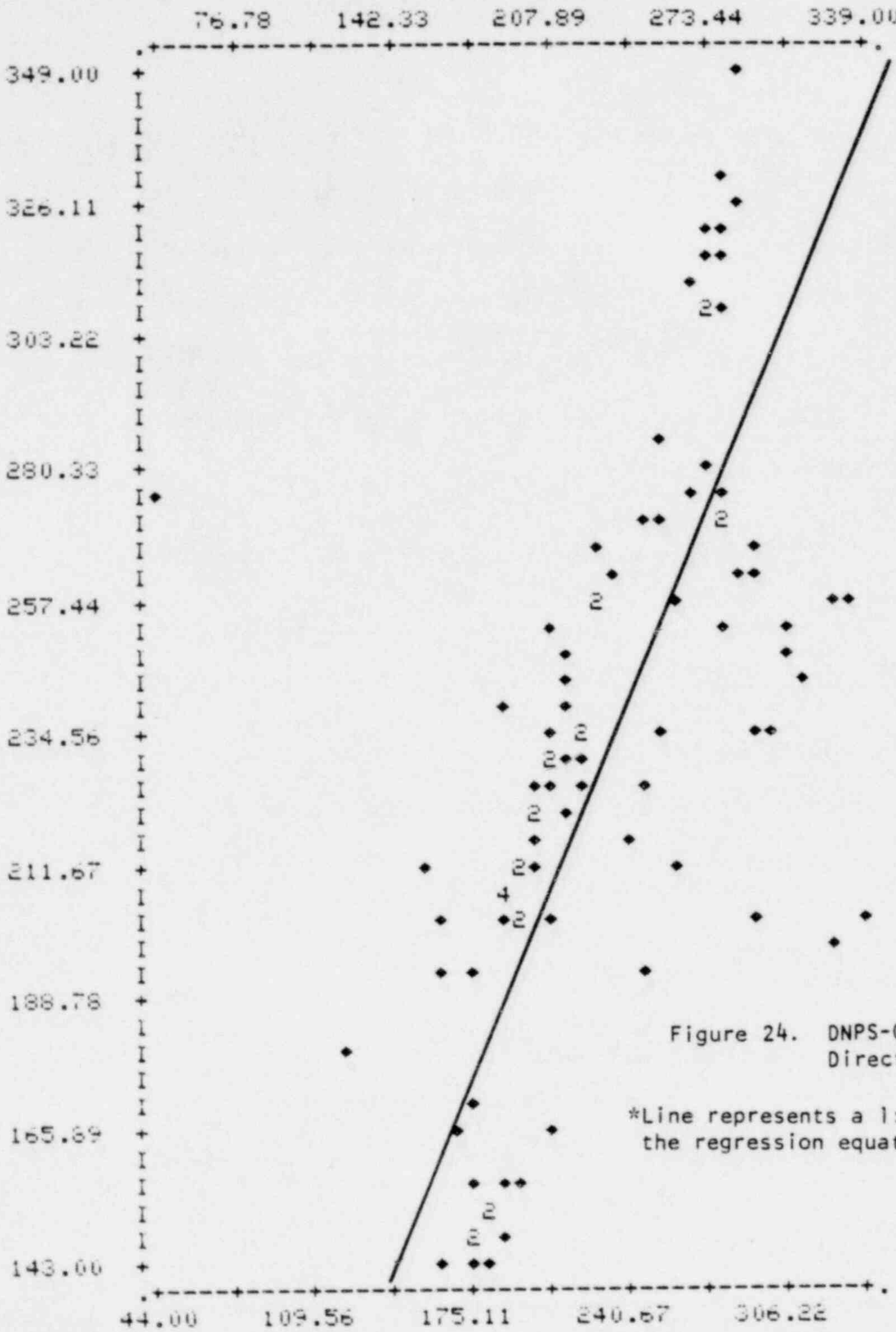


Figure 24. DNPS-QUAD 400 ft. Wind Direction, 4-7 June 1973

*Line represents a 1:1 correspondence, not the regression equation.

CORRELATION (R)-	.58439	R SQUARED -	.34151
SIGNIFICANCE R -	.00001	STD ERR OF EST -	39.91392
INTERCEPT (A) -	108.65060	STD ERROR OF A -	18.75573
SIGNIFICANCE A -	.00001	SLOPE (B) -	.54681
STD ERROR OF B -	.07960	SIGNIFICANCE B -	.00001

PLOTTED VALUES - 93
EXCLUDED VALUES- 0
MISSING VALUES - 3
******** IS PRINTED IF A COEF. CANNOT BE COMPUTED.

Regression equations were generated from the paired data consisting of all the observations on the one hand and the data less the three observations affected by thunderstorms. Then the hourly wind direction values at DNPS on 8 June 1973 were "predicted" from the observed wind at QUAD. In the first case the DNPS directions were "predicted" with a standard error of 8.4 degrees; in the second the standard error was 6.8.

It would appear that additional efforts would be worthwhile.

3.2.2 400 ft. elevation wind speed

The hourly wind speed values were processed in a manner similar to that used on wind direction. The average speed differed by less than one mph. The statistics are summarized in Table 7 .

TABLE 7

Comparative Statistics for DNPS and QUAD
4 June through 7 June 1973
400 ft. Wind Speed (mph)

Name	Value	
	DNPS	QUAD
Mean	15.4	14.6
Standard Deviation	4.9	4.8
Maximum	23	25
Minimum	2	5
Range	21	20
Skewness	-0.517	-0.023
Kurtosis	-0.178	-0.478

(DOWN) DSP400 DNPS 400 FT WIND SPEED JUNE
(ACROSS) QSP400 QUAD 400 FT WIND SPEED JUNE

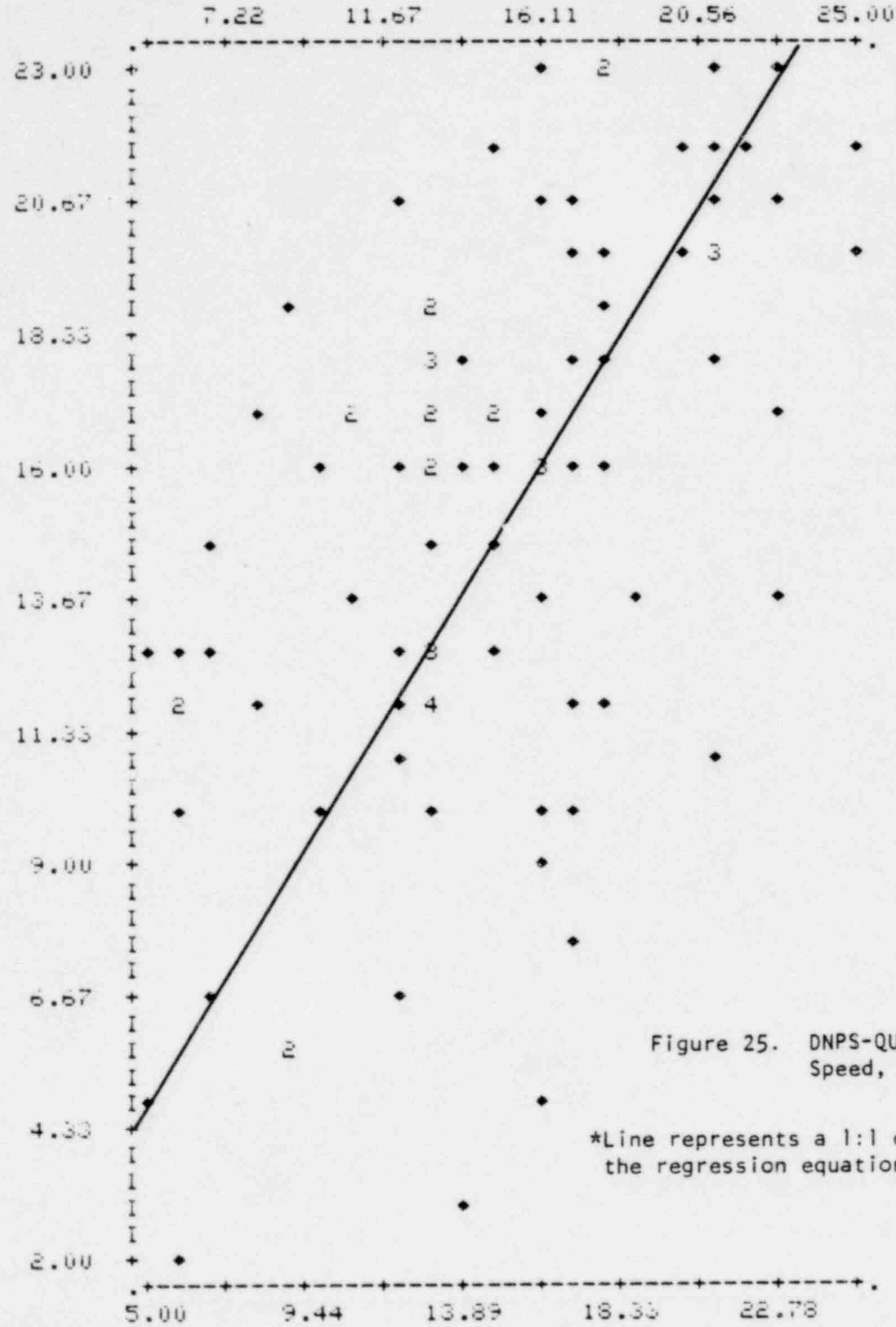


Figure 25. DNPS-QUAD 400 ft. Wind Speed, 4-7 June 1973

*Line represents a 1:1 correspondence, not the regression equation.

CORRELATION (R)-	.54697	R SQUARED -	.29917
SIGNIFICANCE R -	.00001	STD ERR OF EST -	4.12217
INTERCEPT (A) -	7.17710	STD ERROR OF A -	1.36075
SIGNIFICANCE A -	.00001	SLOPE (B) -	.55656
STD ERROR OF B -	.08833	SIGNIFICANCE B -	.00001

PLOTTED VALUES - 95
EXCLUDED VALUES- 0
MISSING VALUES - 1
******** IS PRINTED IF A COEF. CANNOT BE COMPUTED.

The regression equation (figure 25) generated from the 4 - 7 June 1973 data and applied to the 8 June 1973 wind speeds "predicted" the DNPS 400 ft. values with a standard error of 3.7 mph.

3.2.3 400-35 ft. differential temperatures

Hourly differential temperatures were also processed in a manner similar to that described in 3.2.1. The statistics are summarized in Table 8.

TABLE 8

Comparative Statistics for DNPS and QUAD
4 June through 7 June 1973
400-35 ft. Differential Temperature

Name	Value	
	DNPS	QUAD
Mean	0.38	-0.88
Standard Deviation	2.96	1.65
Maximum	9.8	4.1
Minimum	-2.8	-3.1
Range	12.6	7.2
Skewness	1.510	1.188
Kurtosis	1.644	0.514

The regression equation (figure 26) generated from the 4-7 June 1973 data and applied to the 8 June 1973 differential temperatures "predicted" the DNPS values with a standard error of 1.8 degrees. This amounted to a stability class interval standard error of 0.8, or approximately one interval.

(DOWN) DDELTATB DNPS DIFF. TEMP. 400-35 FT JUNE
(ACROSS) QDELTATB QUAD DIFF. TEMP. 400-35 FT JUNE

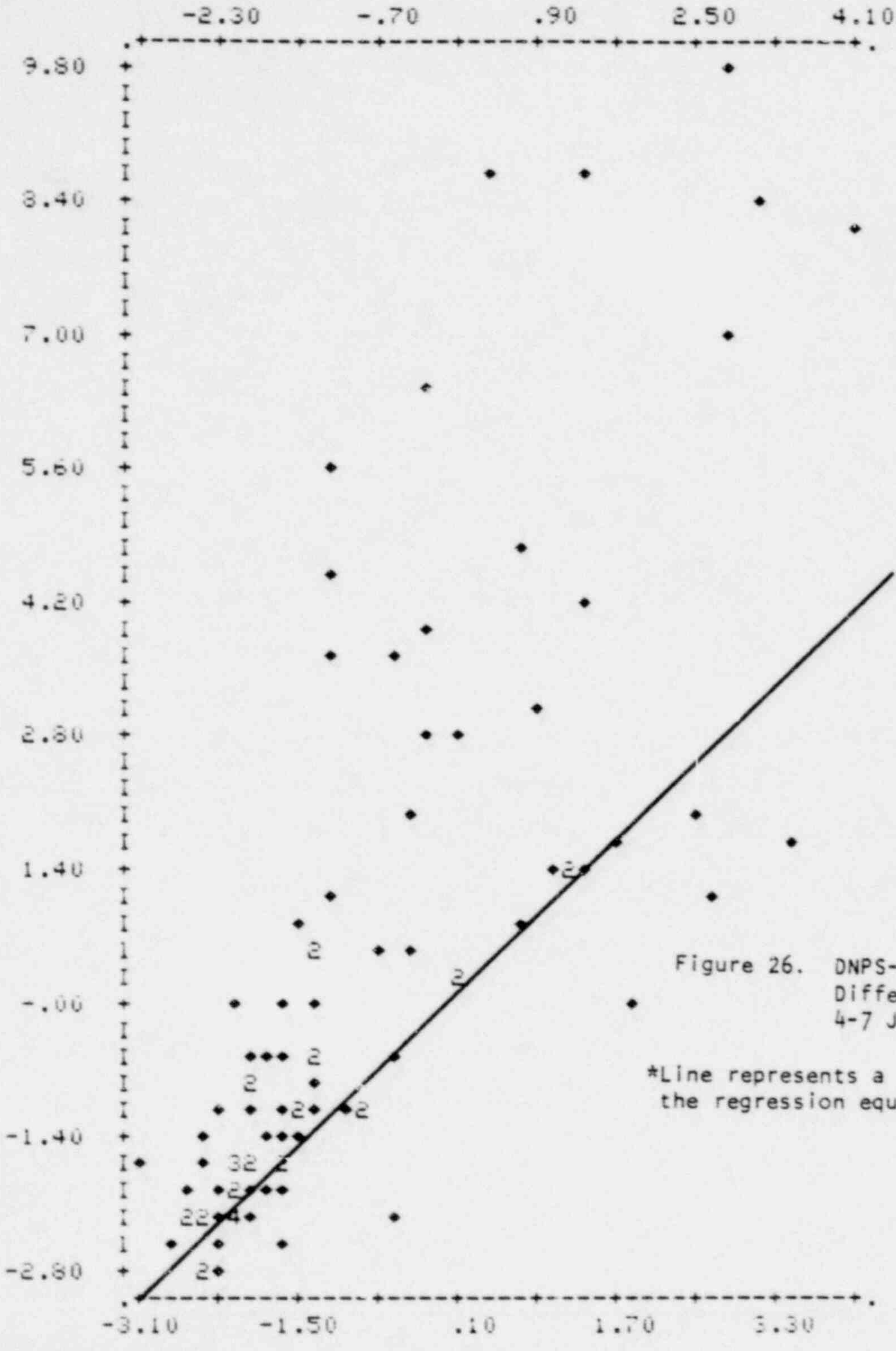


Figure 26. DNPS-QUAD 400-35 ft. Differential Temperatures, 4-7 June 1973

*Line represents a 1:1 correspondence, not the regression equation.

CORRELATION (R)-	.75784	R SQUARED -	.57433
SIGNIFICANCE R -	.00001	STD ERR OF EST -	1.94358
INTERCEPT (A) -	1.65982	STD ERROR OF A -	.23037
SIGNIFICANCE A -	.00001	SLOPE (B) -	1.38167
STD ERROR OF B -	.12401	SIGNIFICANCE B -	.00001

PLOTTED VALUES - 94
EXCLUDED VALUES- 0
MISSING VALUES - 2
******** IS PRINTED IF A COEF. CANNOT BE COMPUTED.

4. Quantitative Relationships Between Measurements at a Single Site

4.1 Wind shear and atmospheric stability

A power law having the form

$$\frac{V_2}{V_1} = \left(\frac{Z_2}{Z_1}\right)^p \qquad (2)$$

was evaluated to generate the exponent "p" using measured hourly wind speeds at the elevations Z_1 and Z_2 where Z_2 is greater than Z_1. The exponents were then grouped according to stability class. The results are given in Table 9. Good agreement was found when the values were compared to those found in the literature. The QUAD and DNPS values are shown in figure 27 along with the recommended ASME values.

These results suggest the possiblity of inferring stability from wind measurements with a reasonable degree of confidence.

TABLE 9

Mean (and standard deviation) "p" values

Station	Elevations	#	E.U.	#	M.U.	#	S.U.	#	N.	#	S.S.	#	M.S.	#	E.S.
									Stability Class						
DNPS	400-35[1]	(0)		(0)		(2)	.19(.07)	(80)	.29(.10)	(63)	.58(.25)	(17)	.83(.22)	(5)	1.10(.19)
DNPS	300-35[1]	(0)		(0)		(2)	.16(.00)	(80)	.26(.12)	(63)	.52(.26)	(17)	.81(.25)	(5)	1.14(.19)
DNPS	125-35[1]	(0)		(0)		(2)	.50(.06)	(80)	.41(.19)	(63)	.65(.38)	(17)	.97(.39)	(5)	1.56(.33)
DNPS	400-125[2]	(0)		(0)		(0)		(124)	.05(.16)	(80)	.44(.24)	(53)	.57(.15)	(7)	.56(.13)
QUAD	400-35[2]	(0)		(0)		(15)	.09(.05)	(145)	.11(.14)	(94)	.31(.14)	(9)	.57(.28)	(0)	
QUAD	300-35[2]	(0)		(0)		(15)	.10(.07)	(148)	.13(.14)	(91)	.30(.18)	(9)	.61(.27)	(0)	
QUAD	125-35[2]	(0)		(0)		(15)	.01(.10)	(148)	.00(.16)	(91)	.13(.21)	(9)	.56(.53)	(0)	
QUAD	400-125[2]	(0)		(0)		(14)	.18(.18)	(146)	.24(.27)	(97)	.51(.33)	(6)	.58(.22)	(0)	
ASME	Guide		.25		.25		.25				.50		.50		.50

[1] For the period 1 June through 7 June 1973

[2] For the period 4 June through 14 June 1973

4.2 Wind Speed and Stability

The DNPS 400 ft. wind speed was "predicted" using the "p" values generated for the 300-125 ft. elevations on 8 June 1973. The standard error of estimate was 4.7 mph. Some negative "p" values occurred because the observed speed at the upper elevation was less than that at the lower elevation. These results suggest the possiblity of inferring wind speed at an uninstrumented greater elevation from wind measurements at lower elevation.

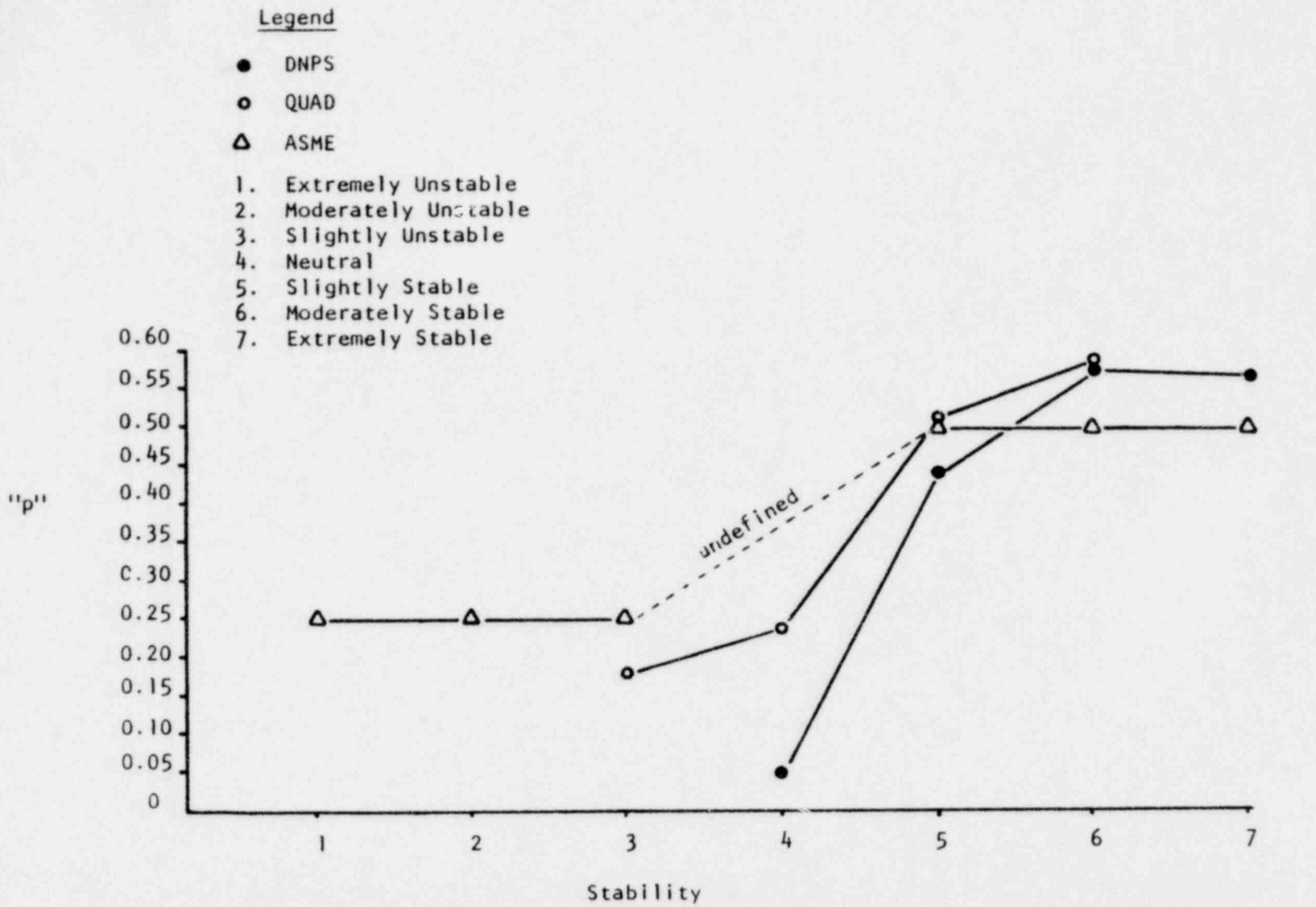


Figure 27. Average "p" Values Based on 400-125 ft. Wind Speed, 4 June through 14 June 1973

Appendix E

Literature on Atmospheric Transport Modeling

Literature on Atmospheric Transport Modeling

1. I. VanDerHoven et al. "Recent Analytical and Experimental Efforts on Single Source Effluent Dispersion to Distances of 100 km", IAEA/SM-181/8, circa 1973.

2. AIF/NESP-007b "Gas Tracer Study of Roof Vent Effluent Diffusion at Millstone Nuclear Power Station", Atmoic Industrial Forum, October 1975.

3. NUREG-0373 "Dispersion in the Wake of a Model Industrial Complex", February 1978.

4. CONF-770901 "Proceedings of a Workshop on the Evaluation of Models Used for the Environmental Assessment of Radionuclide Release", Oak Ridge National Laboratory, April 1978.

5. ORNL-5382 "The Evaluation of Models Used for the Assessment of Radionuclide Releases to the Environment, Progress Report for the Period April 1976 through December 1977", Oak Ridge National Laboratory, June 1978.

6. ORNL/TM-6663 "The Evaluation of Selective Predictive Models and Parameters for the Environmental Transport and Dosimetry of Radionuclides", Oak Ridge National Laboratory, July 1979.

7. NUREG/CR-0798 "Evaluation of Emperical Atmospheric Diffusion Data", October 1979.

8. NUREG/CR-0936 "Recommendations for Meteorological Measurement Programs and Atmospheric Diffusion Prediction Methods for Use at Coastal Nuclear Reactor Sites", October 1979.

9. ORNL-5528 "The Uncertainty Associated with Selected Environmental Transport Models", Oak Ridge National Laboratory, November 1979.

10. NUREG/CR-1286 "Rancho Seco Building Wake Effects on Atmospheric Diffusion: Simulation in a Meteorological Wind Tunnel, February 1980.

Appendix F

Dresden Environmental Dose Pathway Study of Noble Gas Plume Radiation

Section I

DRESDEN EDPS - EXTERNAL DOSE STUDY
(1978-1979)

I. Introduction

The main source of population exposure near nuclear power stations has been gamma radiation from radioactive noble gases. The doses, which are calculated from effluent measurements on the basis of dispersion models (RG 1.109), decrease rapidly as a function of distance from the point of release. Doses have been measured at several installations in the course of special studies, but routine environmental monitoring programs usually report "less than" values because the doses are exceeded by the natural background radiation. Even fluctuations in the natural radiation background can be of the same magnitude as the dose from airborne effluents.

The purpose of the study was to test a procedure for determining external radiation exposure from airborne effluent throughout the area around the station with reasonable effort and cost at levels of 10 mR/year and less. The system consisted of TLD's placed at 16 locations near the station perimeter for measuring the total exposure for 3-month periods, two pressurized argon ionization chambers (PIC's) for distinguishing between the natural radiation background and radiation exposure from airborne effluent, and periodic survey meter readings with a detector sensitive at the μR/hr level at the 16 locations to determine radiation background differences among them. Comparison of measured with computed exposures from airborne effluent at these locations can then be used to calibrate the computational model for predicting exposures at more distant locations.

II. Study Plan

In the 1978-1979 period, Dresden Station was ringed by 16 dosimetric measurement locations at or near the site boundary in the directions used for dispersion calculations (Figure 1 and Table 1). Two of the locations (H and O), in approximately opposite directions, had Pressurized Ionization Chambers (PIC's) for continuously reading (by 10-second integrations) the gamma-ray exposure rate. Thermoluminescent dosimeters (5 per station) were placed at all stations for 3-month exposure integrations. A field technician measured the instantaneous background radiation exposure with a scintillation survey meter calibrated relative to the PICs, at each location, at 2-week intervals, and serviced the PICs. Thus the natural radiation background was recorded continuously (except when the noble gas plume was nearby) by the PICs at two locations, and fortnightly at the other 14 locations. Interpolations based on these two sets of measurements will yield the background exposure of the TLDs

A. PIC Program

Pressurized Ionization Chambers manufactured by Reuter-Stokes, model RSS-111, were used in this program. The RSS-111 Environmental Radiation Monitor is a complete ultra-sensitive gamma exposure monitoring system designed to measure and record the low level exposure rates such as those due to fallout and natural background

TABLE 1

DRESDEN E.D.P.S. MONITORING LOCATIONS

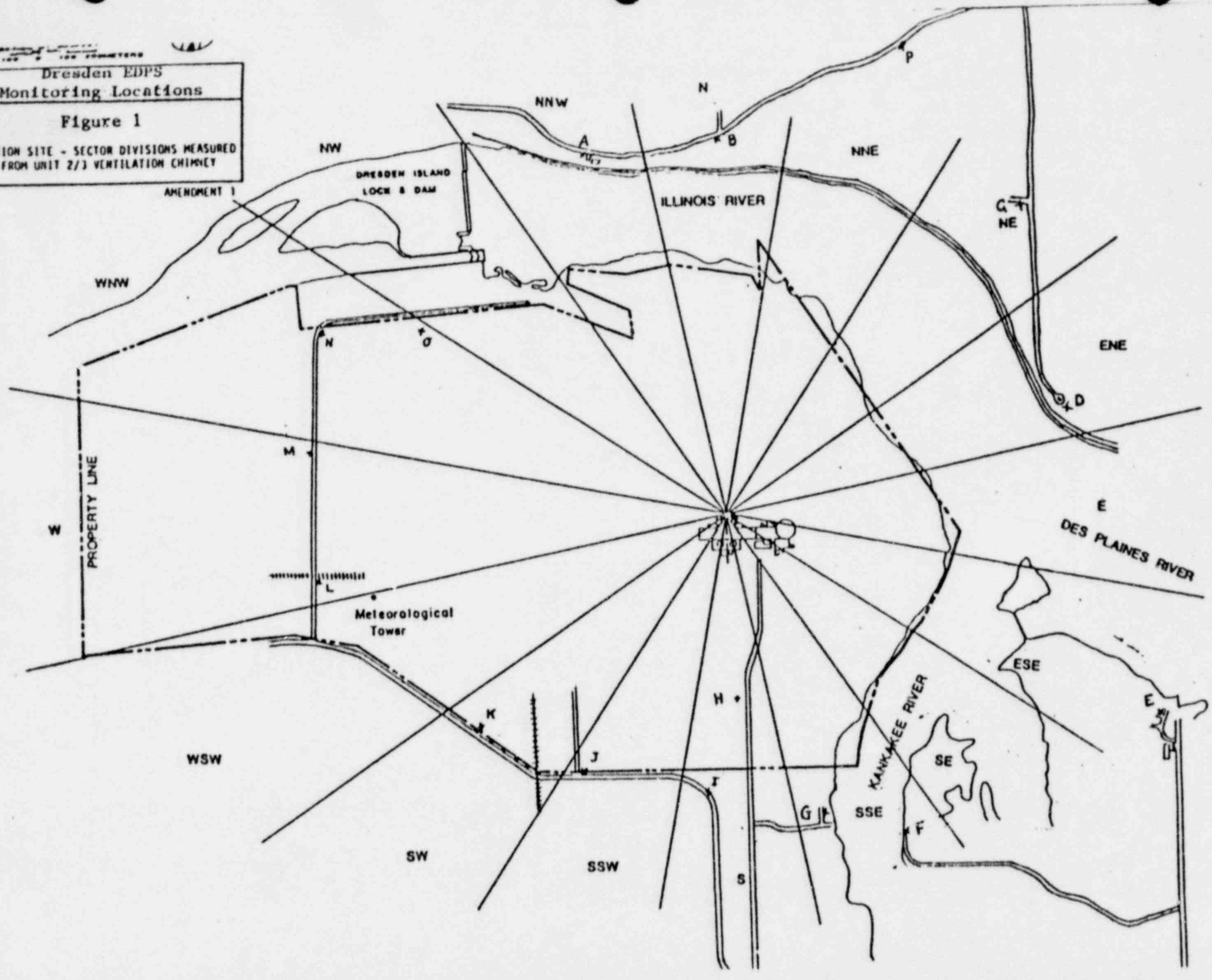
Site	Sector	Direction	Range	Location
A	NNW	340°	4500'	Fence by small tree 60 ft. from end of driveway.
B	N	0	4720'	Fence behind tree by telephone service terminal.
P	NNE	22°	6370'	On telephone service terminal.
C	NE	45°	5440'	Small tree 50 yds. west of fork.
D	ENE	74°	4630'	Air sampling station at Bennett Farm.
E	E	116°	5130'	On fir tree by intersection of driveway.
F	SE	152°	5250'	Pheasant Trail Air Monitoring Station
G	SSE	164°	3750'	On tree by driveway of house.
H*	S	179°	2370'	Onsite No. 3 Air Sampling Station behind shack.
I	S	187°	3220'	On fence by Culvert behind "No Trespassing" sign.
J	SW	211°	3250'	On fence behind "No Trespassing" sign - corner of Met. Tower Road.
K	SW	230°	3830'	On fence by telephone pole, 5th pole past railroad tracks.
L	W	260°	4810'	On fence post by large tree 50 yards before railroad tracks.
M	W	279°	4900'	On small tree 0.4 miles past railroad tracks.
N	WNW	295°	5380'	On small shrub by end of fence post.
O*	NW	302°	4030'	On site No. 1 N.W. corner. Attached to PIC.

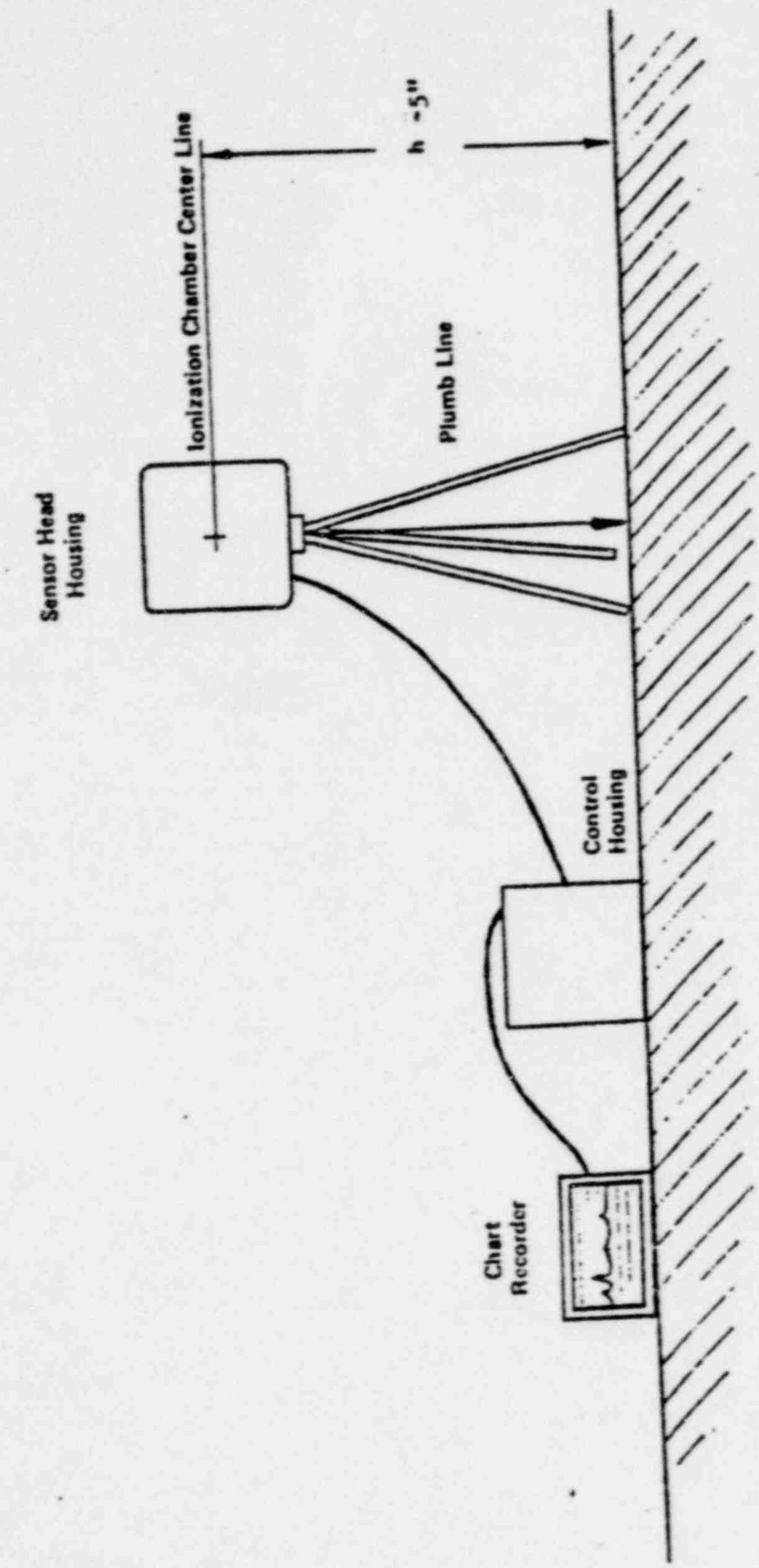
*Pressurized argon ion chamber location.

Dresden EDPS
Monitoring Locations

Figure 1

STATION SITE - SECTOR DIVISIONS MEASURED FROM UNIT 2/3 VENTILATION CHIMNEY





DETECTOR AND READ-OUT SET UP

FIGURE 2

A. PIC Program (continued)

radiation. The RSS-111 is housed in two cable-connected weather-proof enclosures (Figure 2). The sensor housing contains a high pressure spherical ionization chamber with direct mounting to a solid state electrometer. The control housing contains the read-out device. The spherical ionization chamber is filled to a pressure of 25 atmospheres (absolute) with ultra-purity argon. When radiation is incident upon the chamber, the ion pairs produced in the active volume are swept to the electrodes by a collecting potential. The resulting current is measured by an electrometer and can be related directly to the free air exposure rate.

The PICs provide numerous data points to record the exposure rate, so that precise distinctions can be made between the natural background continuum and periodic increases due to station effluents. The background varies gradually at a given location because of the changing accumulation of radon daughters in the ground near the surface and in the ground level air, and because of changes in shielding by snow, rain, and vegetation against radiation emitted by radionuclides in the soil and rock. Changes in the background value observed with the PICs provides adjustments for the background values determined fortnightly with the survey instrument. The survey meter consists of a 2" x 2" NaI (Tl) crystal shielded with cadmium to minimize response to energies below about 80 KeV. The system yields a count rate of about 4000 counts per minute when the pressurized ion chamber indicates a dose rate of about 8 μR/hr. The gamma readings, with adequate intercalibration with the PIC, will provide additional information on the natural background dose rate at the TLD stations and may therefore enable measurement of doses due to station releases at these locations to be calculated.

The computer-processed PIC data consisted of, on an hour-by-hour basis, the average total exposure rate (μR/hr), the standard deviation of the average, the measured (or assumed) background, and the estimated plume contributions to the total exposure rate (the difference between the total and background). (An "assumed" background rate is the last total exposure rate measured without a plume present. A plume was considered present if the total exposure rate exceeded the background by 3σ.) The hourly plume contributions were summed to give the monthly measured total. This total was then compared to calculated exposure rates computed with accepted sector-averaged dispersion models (RG 1.109) and station reported gross radioactivity levels. (The gross radioactivity was assumed to have a 0.8 MeV average energy because the predominant source was Dresden 1 steam jet air ejector offgas, with a nominal one-hour holdup for decay. Table 2.) In addition, at all 16 locations, the sector-averaged dispersion model, averaged monthly release rates, and joint frequency wind roses were used to determine exposures.

The PIC readings are based on calibration by the manufacturer. The calibration is in terms of Ra-226. The radiation exposure response is energy dependent : it is within 3 percent of the Ra-226 value from 0.5 to 2.5 MeV, but the response factor is higher above 2.5 and below 0.5 MeV, especially at 0.1 MeV. The PIC does not detect external alpha and beta particles or gamma rays below 0.06 MeV.

B. Thermoluminiscent Dosimeter Readings

Lithium Fluoride thermoluminiscent dosimeters (TLDs) were emplaced at 16 locations surrounding the plant, approximately one location per sector used for dose calculations. Locations are given in Table 1 and Figure 1. TLD badges remained on station for three months. Exposure rates were measured and compared with those calculated from release and meteorological data, gamma survey data (see Section C), and where possible, from PIC data (see Section A).

Several different TLD packagings were used in the program. They are:

(a) Al Shield in PVC holder (~325 mg/cm²)

(b) Al Shield in Polyethylene Vial holder (~325 mg/cm²)

(c) Polyethylene Shield in PVC holder (~56 mg/cm²)

(d) Polyethylene Shield in Polyethylene Vial holder (~56 mg/cm²)

(e) PVC Sphere Isotropic holder (~1 gm/cm²).

The comparative data for various packaging holders are summarized in Section III-B. The Polyethylene Shield in Polyethylene Vial (~56 mg/cm²) and PVC sphere isotropic holders (~1 gm/cm²) were chosen for the period of August 23, 1978, through March 14, 1979.

The TLDs are calibrated at frequent intervals and are found to be accurate within a standard deviation of ±10%.

C. Survey (gamma) Meter Readings

Gamma survey readings were made at each of the TLD stations on alternate weeks using a survey meter with a 2" x 2" NaI (Tl) crystal shielded with cadmium to minimize response to energies below about 80 KeV. The results are summarized in Section III-C. The following equation was used to relate counts per minute to μR/hr:

$$\mu R/hr = 8.4\times10^{-4}(cpm)+4.1.$$

III. Results

A. PIC Data*

Because of the operational difficulties with the PICs, useful data were not collected until August, 1978. Good data collection continued through the fall when, in November, Dresden 1 was shut down for an extended outage. The PICs continued to gather data through

*The PICs were calibrated by the manufacturer and as discussed later in the text, may measure high by a factor of 1.2 when compared to TLDs.

III-A. PIC Data (continued)

February, 1979, but during this period the Station release rates were so low (Table 2) that the measured plume exposures were considered to be in the error range of background (with the standard deviation of background averaging, between 0.1 - 0.2 μR/hr, 2σ over a month's time is 140 - 200 μR). For this reason, most data analyses were performed only on the August - October data sets.

Table 3 presents the calculated sector averaged exposures for the May, 1978 - February, 1979 period. For the August - October 1978, period, the following measured vs. sector-averaged calculated exposures were found:

	Dose (μR)			
	Site O		Site H	
	Measured	Calculated	Measured	Calculated
August	2548	1419	3723	3285
September	1130	1062	2947	1283
October	514	615	3465	1660

In five of these six cases, the measured data exceeded the value calculated using the joint frequency wind roses.

Using the hourly data summed to give the monthly total at location O, the computed values were 1487, 821, and 249; at H, 2983, 1307, and 1651 μR per month for August - October. In all six of these cases, the measured exceeded the model value. The effect, the measured exceeding the model dose, is believed due to three factors:

* * * * * * * * *

Table 2

Gross Radioactivity Release Rates (mCi/sec)*

Month	Dresden 1	Dresden 2/3	
	Chimney	Chimney	Reactor Bldg. Vent
1978			
May	21.4	3.5	0.06
June	34.3	1.3	0.04
July	37.1	2.3	0.01
August	45.2	0.5	0.01
September	42.3	1.4	0.05
October	49.3	0.9	0.01
November	0.02	0.7	0.01
December	0.004	0.5	0.02
1979			
January	0	0.1	0.02
February	0.003	0.6	0.02

* An assumed average gamma energy of 0.8 MeV and no radiological decay were used in the calculation.

TABLE 3

CALCULATED MONTHLY PLUME EXPOSURES (mR)
(Sector Averaged Model)

Site	Range-M	Sector	05/78 Dose	06/78 Dose	07/78 Dose	08/78 Dose	09/78 Dose
A	1371.0	NNW	1.678E 00	2.608E 00	2.349E 00	4.390E 00	1.764E 00
B	1438.0	N	6.351E-01	1.967E 00	2.908E 00	5.240E 00	1.913E 00
P	1941.0	NNE	1.193E 00	1.636E 00	2.600E 00	3.241E 00	2.541E 00
C	1657.0	NE	1.397E 00	1.566E 00	2.556E 00	3.646E 00	4.638E 00
D	1410.0	ENE	1.321E 00	1.715E 00	2.572E 00	2.762E 00	2.958E 00
E	1563.0	E	3.864E-01	7.210E-01	1.041E 00	1.877E 00	1.085E 00
F	1599.0	SE	5.734E-01	1.166E 00	7.501E-01	1.728E 00	1.624E-01
G	1142.0	SSE	7.732E-01	1.578E 00	1.023E 00	2.304E 00	2.206E-01
H*	722.0	S	1.091E 00	9.239E-01	2.621E 00	3.285E 00	1.283E 00
I	981.0	SSW	7.812E-01	6.943E-01	1.963E 00	2.412E 00	8.706E-01
J	990.0	SW	8.075E-01	1.512E 00	2.233E 00	1.328E 00	8.954E-01
K	1167.0	WSW	1.018E 00	1.115E 00	2.224E 00	1.281E 00	8.586E-01
L	1465.0	W	9.789E-01	9.161E-01	1.776E 00	1.010E 00	8.450E-01
M	1493.0	WNW	9.605E-01	8.989E-01	1.743E 00	9.911E-01	8.292E-01
N	1639.0	WNW	7.236E-01	6.385E-01	1.601E 00	1.106E 00	8.665E-01
O*	1228.0	NW	9.223E-01	8.116E-01	2.061E 00	1.419E 00	1.062E 00

Site	Range-M	Sector	10/78 Dose	11/78 Dose	12/78 Dose	01/79 Dose	02/79 Dose
A	1371.0	NNW	1.167E 00	1.226E-02	2.628E-02	6.837E-04	3.125E-02
B	1438.0	N	2.546E 00	1.754E-02	2.167E-02	3.319E-04	2.733E-02
P	1941.0	NNE	3.319E 00	2.250E-02	2.250E-02	1.400E-03	6.868E-03
C	1657.0	NE	3.260E 00	2.270E-02	4.468E-02	1.040E-03	7.317E-03
D	1410.0	ENE	1.977E 00	3.209E-02	2.892E-02	3.505E-03	2.047E-02
E	1563.0	E	1.659E 00	2.305E-02	6.601E-03	1.015E-02	2.708E-02
F	1599.0	SE	1.339E 00	1.315E-02	2.968E-03	1.597E-03	1.437E-02
G	1142.0	SSE	1.786E 00	1.804E-02	4.106E-03	2.250E-03	1.975E-02
H*	722.0	S	1.660E 00	5.310E-02	4.031E-03	6.405E-04	3.017E-02
I	981.0	SSW	1.273E 00	3.867E-02	2.902E-03	4.670E-04	2.251E-02
J	990.0	SW	1.713E 00	3.158E-02	2.935E-03	2.125E-03	1.800E-02
K	1167.0	WSW	6.894E-01	1.659E-02	1.364E-03	2.873E-03	4.370E-03
L	1465.0	W	4.594E-01	7.374E-03	7.265E-03	8.621E-04	2.145E-02
M	1493.0	WNW	4.508E-01	7.235E-03	7.128E-03	8.459E-04	2.104E-02
N	1639.0	WNW	4.798E-01	5.952E-03	9.752E-03	1.006E-03	1.543E-02
O*	1228.0	NW	6.149E-01	7.826E-03	1.281E-02	1.348E-03	2.033E-02

*Sites H and O had a PIC.

(1) plume transfer from the metro tower - indicated sector into another sector as the plume travels downwind, (2) the failure of the model to account for measured radiation which originated in adjacent sectors, and (3) the sector averaging concept itself which creates a "plume" of concentration lower than that actually found. The long relaxation length of gamma radiation in the air means that radiation can travel across sectors to impact on the PICs.

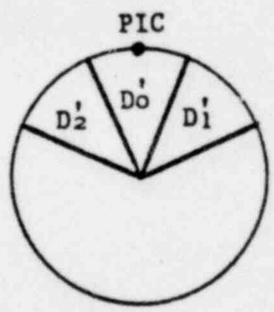
The first factor, plume transfer into and out of sectors, is not thought to be as important as the other factors because in an area such as Illinois, with a rather uniform wind pattern, the probability of transfer out of one sector is nearly the same as the probability of transfer into a sector.

Two modeling attempts were then tried in order to adjust the data of sectors O and H to account for radiation originating in other sectors. The first model was a simple one, it assumed that all radiation from plumes in a PIC sector was detected along with a portion of radiation from plumes in adjacent sectors. The second model was more complicated, it involved a computer-calculated exposure based on the actual direction of the plume and its relationship to the direction and range of the PICs.

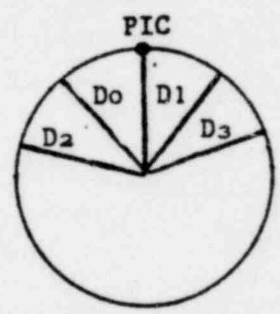
Model 1

In the simple model, two cases were envisioned: Case A placed the PIC in the center of a three sector system. In case B, the PIC was placed in the center of a four sector system on the boundary between the two inner sectors.

Case A



Case B



For case A (Table 4), the most commonly used sector arranged model, Dm, the PIC measured exposure, is represented by:

> $Dm = f(\Phi)\, D'_o + f(1)(D'_1 + D'_2)$ with $f(\Phi) \equiv 1$; and $f(1)$ is a fraction <1. (The value in the parenthesis is the number of sector widths between the PIC and each sector's centerline.)
>
> $Dm = D'_o + (0.34 \pm 0.76)(D'_2 + D'_1)$

For Case B (Table 5):

$$Dm = f(\tfrac{1}{2})(Do + D1) + f(1\tfrac{1}{2})(D2 + D3)$$

$$= (0.67 \pm 0.56)(Do + D1) + (0.13 \pm 0.36)(D2 + D3)$$

III-A. PIC Data (continued)

Table 4

Case A Model (μR)

	Dm	D_2'	D_o'	D_1'	f(o)	f(1)
Site O						
August	2548	619	1487	2809	1.0	0.29
September	1132	1081	821	748	1.0	0.17
October	519	722	249	826	1.0	0.08
Site H						
August	3723	1935	2983	3374	1.0	0.14
September	2947	1187	1307	393	1.0	0.96
October	3465	1942	1651	2587	1.0	0.41

$Dm = f(\phi)\,D_o' + f(1)(D_2' + D_1')$

$= D_o' + (0.34 \pm 0.76)(D_2' + D_1')$ where the error is $\pm 2.52\sigma$.

* * * * * * * *

Table 5

Case B Model (μR)

	Dm	D_2	D_o	D_1	D_3*	f(½)	f(1½)
Site O							
August	2548	959	905	2432	1432	0.68	0.09
September	1130	921	1033	634	863	0.66	0.00
October	514	488	645	620	584	0.32	0.11
Site H							
August	3723	1174	2166	3921	2426	0.53	0.14
September	2947	1271	1161	801	1078	1.00	0.42
October	3465	1442	1835	2478	1918	0.80	0.00

* Computed from $(D_2 + D_o + D_1) \div 3$

$Dm = f(\tfrac{1}{2})(D_o + D_1) + f(1\tfrac{1}{2})(D_2 + D_3)$

$= (0.67 \pm 0.56)(D_o + D_1) + (0.13 \pm 0.36)(D_2 + D_3)$ where the error is $\pm 2.57\sigma$.

Because there were so few data points - six cases for determining each fraction - the error at the 95% confidence level is quite large. Nevertheless, both Cases A and B show that the most commonly used finite plume model, with no radiation contributions considered from adjacent sectors, underestimates the actual plume exposure by approximately 60%.

Tables 6-A and 6-B summarize the PIC data on a weekly basis for the period August 1978 - March 1979.

Table 7 summarizes the PIC background data for the same period of time.

Model 2

In the more complicated, off-axis model, the plume model from Eq.7.43 of Meteorology and Atomic Energy was used. Hourly emissions, determined by assuming the daily reported emissions were uniform throughout the day, and off-axis distances from the monitor to the plume center line were accounted for in the model. The plume center line was considered to be the down wind direction determined from the wind measured at 90m above ground.

The cumulative hourly doses for the three months are summarized in Table 8. As reported previously, the sector model consistently under-calculated the doses indicated in Column (5). The off-axis plume model, on the other hand, over-calculated on four out of the six cases (Table 8, Column 7). The high doses modeled at Site "H" in August were due largely to four days (the 3rd, 4th, 29th and 31st). These were days when the downwind direction was close to the monitor bearing and differential temperatures were in the "extremely unstable" class. Without these days, the measured and modeled doses for the month were 1735 and2565 urad, which gives a calculated to measured ratio of 1.48. The average calculated to measured ratio is 1.2 for the six measurements listed in Column 7 of Table 8 when the data for August at station "H" are adjusted.*

It is concluded that application of an off-axis plume model to compute gamma air doses can be realistic and useful. However, the model can give unrealistically high values under conditions of strong temperature lapse rates. Examination of the data indicates that the model can give more accurate results if atmospheric stability classes are constrained to the neutral and stable categories. Furthermore, the analysis indicates that wind direction measured at an intermediate height (46m in this case) may better represent the plume center line during extremely unstable atmospheric conditions.

B. Thermoluminiscent Dosimeter and Gamma Survey Data

Table 9 summarizes the data, on comparison of dose with TLDs in various packing materials.

* See ** footnote to Table 8. The actual difference may be 1.4.

III-A. PIC Data (continued)

Table 6-A

Summary of PIC Data from August 1978 - March 1979

ON-SITE #1 LOCATION - 0
(weekly average)

Dates	Hours	ΣμR Gross	ΣμR Bkgd	Ave. μR/hr Total	Ave. μR/hr Bkgd	Ave. μR/hr Net
1978						
07/24-07/31	170	1704	1491	10.0	8.8	1.2
07/31-08/09	216	2292	1941	10.6	9.0	0.4
08/09-08/16	169	3170	1567	18.9	9.3	9.6
08/16-08/23	170	1692	1547	10.0	9.1	0.9
08/23-09/04	285	3045	2597	10.7	9.1	1.6
09/04-09/10	147	1539	1398	10.5	9.5	1.0
09/10-09/16	144	1704	1435	11.2	9.4	1.8
09/16-09/24	192	2087	1670	10.9	8.7	2.2
09/24-10/01	169	1844	1500	10.9	8.9	2.0
10/01-10/10	211	2019	1849	9.6	8.8	0.8
10/10-10/15	121	1093	1086	9.0	9.0	0.0
10/15-10/22	169	1557	1520	9.2	9.0	0.2
10/22-10/29	167	1755	1468	10.5	8.8	1.7
10/29-11/05	165	1508	1495	9.1	9.1	0.0
11/05-11/12	166	1490	1467	9.0	8.8	0.2
11/12-11/19	170	1522	1488	9.0	8.8	0.2
11/19-11/26	169	1489	1467	8.8	8.7	0.1
11/26-12/03	168	1466	1422	8.7	8.5	0.2
12/03-12/10	167	1421	1380	8.5	8.3	0.2
12/10-12/17	169	1350	1350	8.0	8.0	0.0
12/17-12/24	165	1355	1339	8.2	8.1	0.1
12/24-12/30	145	1207	1189	8.3	8.2	0.1
1979						
12/30-01/06	170	1255	1235	7.4	7.3	0.1
01/06-01/20	323	2264	2260	7.0	7.0	0.0
01/21-01/28	171	1099	1092	6.4	6.4	0.0
01/28-02/04	168	1064	1065	6.3	6.3	0.0
02/04-02/11	167	1019	1019	6.1	6.1	0.0
02/11-02/18	165	970	966	5.9	5.9	0.0
02/18-02/25	171	1029	1014	6.0	5.9	0.1
02/25-03/03	144	874	862	6.1	6.0	0.1
03/03-03/10	170	1199	1192	7.1	7.0	0.1

III-A. PIC Data (continued)

Table 6-B

Summary of PIC Data from August 1978 - March 1979

ON-SITE #3 LOCATION - H
(weekly Average)

Dates	Hours	ΣμR Gross	ΣμR Bkgd	Ave. μR/hr Total	Ave. μR/hr Bkgd	Ave. μR/hr Net
1978						
07/24-07/31	170	2132	1484	12.5	8.7	3.8
07/31-08/09	216	4040	1923	18.7	8.9	9.8
08/09-08/16	169	1786	1526	10.6	9.0	1.6
08/16-08/23	170	1645	1530	9.7	9.0	0.7
08/23-09/04	285	4673	2576	16.4	9.0	7.4
09/04-09/10	147	1510	1389	10.3	9.4	0.9
09/10-09/16	144	1572	1323	10.9	9.2	1.7
09/16-09/24	192	2864	1638	14.9	8.5	6.4
09/24-10/01	169	2203	1468	13.0	8.7	4.3
10/01-10/10	211	1871	1803	8.9	8.5	0.4
10/10-10/15	121	1782	1083	14.7	9.0	5.7
10/15-10/22	169	1970	1508	11.7	8.9	2.8
10/22-10/29	168	3480	1446	20.7	8.6	12.1
10/29-11/05	168	1676	1473	10.0	8.8	1.2
11/05-11/12	166	1448	1421	8.7	8.6	0.1
11/12-11/19	170	1478	1450	8.7	8.5	0.2
11/19-11/26	169	1472	1444	8.7	8.5	0.2
11/26-12/03	168	1456	1408	8.7	8.4	0.3
12/03-12/10	167	1363	1354	8.2	8.1	0.1
12/10-12/17	170	1374	1370	8.1	8.1	0.0
12/17-12/24	165	1397	1379	8.5	8.4	0.1
12/24-12/30	145	1218	1197	8.4	8.3	0.1
1979						
12/30-01/06	172	1300	1280	7.6	7.4	0.2
01/06-01/13	Missing - Due to tape problems					
01/13-01/20	167	1148	1141	6.9	6.8	0.1
01/20-01/28	170	1108	1089	6.5	6.4	0.1
01/28-02/04	170	1104	1104	6.5	6.5	0.0
02/04-02/11	166	1056	1053	6.4	6.3	0.1
02/11-02/18	167	1010	995	6.0	6.0	0.0
02/18-02/25	170	1038	1033	6.1	6.1	0.0
02/25-03/03	144	888	883	6.1	6.1	0.0
03/03-03/10	167	1220	1222	7.3	7.3	0.0
03/10-03/14	96	695	695	7.2	7.2	0.0

III-A. PIC Data (continued)

Table 7

Summary of PIC Background Date μR/hr
(integrated over 4-week intervals)

	Site O	Site H
1978		
August	9.08	8.99
September	9.22	9.02
October	8.88	8.75
November	8.86	8.62
December	8.35	8.27
1979		
January	6.9	6.9
February	6.0	6.1
March	6.20	6.37

* * * * *

Table 8

Cumulative Hourly Gamma Dose (Microrads) - Dresden Nuclear Power Station
August - September 1978

(1) Month	(2) Site	(3) Measured	(4)Δ Sector-Ave'd Model	(5) (4) ÷ (3)	(6) Off-axis Plume Model	(7)** (6) ÷ (3)
August	"O"	2548	1487	0.58	3776	1.48
	"H"	3723	2983	0.80	7219	1.94*
September	"O"	1130	821	0.73	1659	1.47
	"H"	2947	1307	0.44	2198	0.75
October	"O"	514	249	0.48	553	1.08
	"H"	3465	1651	0.48	3429	0.99

Δ Same as Do^1 in Table 4.

* 1.48 after adjustment for periods of extremely unstable weather conditions with downwind direction close to the monitor bearing. See text.

** A portion of this difference between calculated and PIC measured exposures may be due to the PIC calibration techniques. As discussed in in text, the PIC values may be high by a factor of 1.2. If this factor is correct, the difference between the measured and computer-calculated exposures is 40%.

III-B. Thermoluminiscent Dosimeter and Gamma Survey Data (continued)

Table 9

Dose Rates in μR/hr (April 26 - May 19, 1978)

Packaging Material	Locations (±1σ)			
	A	D	F	H
Al Shield, PVC holder	8.40±0.43	7.27±0.24	7.37±0.34	7.84±0.17
Al Shield, Polyethylene holder	8.67±0.35	7.00±0.44	7.39±0.26	8.19±0.67
Polyethylene Shield, PVC holder	9.88±0.17	9.36±0.35	-	9.40±0.29
Polyethylene Shield, Polyethylene Vial holder	-	-	-	9.45±0.75
Isotropic PVC Sphere	7.69±0.26	6.24±0.29	6.27±0.22	6.67±0.61

* * * * * * * *

The TLD packings Polyethylene Shield in polyethylene vial (~56 mg/cm² thickness), and Isotropic PVC holder (~1 g/cm² thickness) were chosen for the study.

Tables 10 and 11 summarize the Gamma Dose Rates measured using TLDs from May 17, 1978, to March 14, 1979.

Comparison of TLD data with PIC data is presented in Table 12. From this the correction factors are obtained to correlate TLD and PIC data.

Table 13 presents the background data obtained from PIC and γ-survey measurements. The γ-survey readings were converted to μR/hr, using a calibration curve between γ cpm and PIC μR/hr.

Table 14 presents the average background from γ-survey readings for all locations except locations H and O (which are the PIC locations - see Table 13).

The differences between the measured background exposure rates and the average of the values at H and O were calculated. Values on 09/10 were omitted because of the apparent error in SM measurements indicated for Table 13.

The following values were also omitted because the plume from the chimney may have been at the location: C, 08/16, 10/22, 11/19; D, 10/08; L, 02/18, 03/03; M, 02/18, 03/03. The averages of the acceptable difference values were all within the standard deviations except the following:

Location	Difference from H and O
E	-1.1 ± 1.0 μR/hr
F	-1.4 ± 0.8 μR/hr
G	-1.1 ± 1.0 μR/hr

III-B. Thermoluminiscent Dosimeter and Gamma Survey Data (continued)

Table 10

Average Environmental Gamma Dose Rates using TLD's in μR/hr (±1σ) (Polyethylene shield in Polyethylene vial holder - 56 mg/cm^2 thickness).

Station	17 May-22 Aug.	23 Aug.-15 Nov.	15 Nov.-14 Mar.
A	11.9±0.2	11.8±0.9	7.4±0.5
B	15.3±0.5	20.3±1.1	9.9±0.7
C	14.3±1.1	18.4±0.7	7.4±0.8
D	12.1±0.2	16.4±1.1	9.4±0.5
E	8.3±0.3	11.4±0.2	6.9±0.4
F	9.5±0.4	14.3±1.8	5.8±0.3
G	9.3±0.3	Missing	Missing
H	10.9±0.5	14.8±0.7	9.3±0.5
I	9.6±0.0	14.8±0.7 (1)	8.8±0.5
J	10.5±0.1	Missing	8.0±0.4
K	10.5±0.1	18.4±0.5	8.3±0.6
L	9.8±0.0	13.4±0.7	9.6±0.6
M	8.2±0.2	10.9±0.2	8.1±0.4
N	8.2±0.2	9.8±0.2	7.5±0.7
O	8.8±0.2	10.9±0.7	7.2±0.4
P	11.8±0.7	16.6±0.2	8.1±0.7

(1) Second badge=15.5±0.5

* * * * *

Table 11

Average Environmental Gamma Dose Rates using TLD's in μR/hr (±1σ) (Isotropic PVC Sphere holder - 1 g/cm^2 thickness).

Station	17 May-22 Aug.	23 Aug.-15 Nov.	15 Nov.-14 Mar.
A		8.9±0.5	5.6±0.3
B		13.9±1.0	6.6±0.9
C		12.4±1.0	5.9±0.3
D	8.3±0.2	11.4±0.5	7.0±0.5
E		9.3±0.4	5.3±0.3
F		9.3±0.4	4.6±0.5
G		Missing	5.6±0.3
H		10.9±1.0	6.1±0.7
I		11.4±0.0	6.9±0.4
J		10.4±1.0	6.6±0.4
K		10.9±1.0	6.4±0.3
L		10.4±1.0	7.4±0.4
M		8.4±1.0	6.1±0.3
N		7.4±0.0	5.3±0.3
O	6.9±0.2	8.4±0.5	6.1±0.4
P		12.4±1.0	6.1±0.5

III-B. Thermoluminiscent Dosimeter and Gamma Survey Data (continued)

Even these are highly uncertain, but this result suggests that there is a consistently lower background toward the southeast. On this basis, the background at all stations except these three was taken to be identical to the averages at O and H, while the background at E, F, and G was taken to be lower by the average value of 1.2 μR/hr.

The typical standard deviation in the differences between the SM measurement at a location and the value at H and O was 0.9 μR/hr, which is larger than desirable in a correction factor. More work needs to be done on this aspect of the project, so that a more consistent correction factor can be applied to the background measurement.

The radiation exposure from airborne effluent given in Table 15 was calculated as follows: Exposure rate values were converted to TLD from PIC values, based on the belief that the TLDs are reliably calibrated but that the PIC calibration is only approximate. In the future, the two systems should be cross-calibrated. Hence, background values from the PIC (see footnote 1 to Table 15) were divided by 1.22, the average ratio from Table 12. Only data from Table 11 were used because these TLDs resemble the PIC in responding mostly to gamma rays. The higher values in Table 10 presumably are due to beta particles and weak gamma rays. These TLDs are a useful indicator of this additional component but cannot be compared to PICs.

The PIC background for TLDs was used at all locations where the SM values agree within 1σ. Only at locations E, F, and G was a different background used. This latter background was an average value for the three locations because of their proximity and the large uncertainty of each set of values.

The measured exposure rates due to airborne effluents in the period November 15 - March 14, average 0.3 ± 0.5 μR/hr (1σ). The average value is consistent with the dispersion calculation for that period of <0.1 μR/hr in Table 3 and the measured values of 0 - 0.2 μR/hr in Tables 6A and 6B at locations H and O. This yields a 3σ minimum detectable level of 1.5 μR/hr or 3 mR per quarter. The standard deviation of 0.6 μR/hr can be inferred to result from the uncertainty of the total measurement, which averages 0.4 μR/hr in Table 11 and of the background value subtracted from it, which appears to have a similar average standard deviation.

Based on the above discussion, the standard deviation of the August 23 - November 15 measured exposure rates due to airborne effluents in Table 15, is somewhat larger than 1 mR, and the MDL is somewhat larger than 3 mR. That suggests that values at all locations except A, M, N, and O can be compared to values computed from dispersion calculations. At an estimate, it appears that the dispersion calculations yield <3mR for A, M, N, and O also, and are within 2 mR of measured values at C, D, and F. Other calculated values appear to be significantly less than the measured average.

The minimum detectable annual exposure by this procedure for four quarterly TLD measurements is $3(4)^{0.5} = 6$mR/year.

The following were taken from Table 15 for the TLD in 1 -g/cm^2

Table 12

Comparison of TLD and PIC Data (μR/hr)

Date	Site	(±1σ) TLD Data 56 mg/cm²	1 g/cm²	PIC Data	Correction Factors (Compared to PIC) 56 mg/cm²	1 g/cm²
08/23/78-11/15/78	H	14.8±0.3	10.9±0.3	13.6	0.92	1.25
08/23/78-11/15/78	O	10.9±0.1	8.4±0.3	10.4	0.95	1.24
11/15/78-03/14/79	H	9.3±0.5	6.1±0.7	7.32	0.79	1.20
11/15/78-03/14/79	O	7.2±0.4	6.1±0.4	7.24	1.01	1.19

* * * * *

Table 13

Comparisons of Data Obtained (instantaneous readings) from PICs and NaI (Tl) γ Survey

	μR/hr Site H		Site O	
Date	γ	PIC	γ	PIC
08/23/78	9.2	9.4	9.4	9.3
09/10/78	10.8	8.5	12.0	9.6
09/24/78	9.0	9.5	9.1	8.9
10/08/78	9.5	8.8	8.8	8.5
10/22/78	8.8	8.8	9.2	8.8
11/05/78	8.4	9.4	8.7	9.0
11/19/78	8.4	8.7	8.9	8.8
12/03/78	8.4	8.0	8.4	8.6
02/04/79	7.0	7.1	6.5	6.2
02/18/79	6.8	6.3	7.1	6.5
Average (±1σ)	8.6±1.2	8.5±1.1	8.8±1.5	8.4±1.1

* * * * *

Table 14

Average Background Dose from γ-Survey

Date	Wind Direction	μR/hr													
		A	B	C	D	E	F	G	I	J	K	L	M	N	P
07/24/78	ESE	8.8	9.6	8.8	9.6	8.4	8.0	8.0	9.2	10.2	9.6	8.6	9.2	P	9.6
07/31/78	W	9.0	9.4	9.2	9.6	8.6	8.4	8.6	9.4	10.6	10.0	9.6	8.4	8.0	9.8
08/16/78	SW	8.6	9.6	9.4	P	9.2	8.4	8.4	9.4	11.4	9.2	9.4	8.6	8.0	10.0
08/23/78	S	9.2	10.8(P)	P	P	9.4	8.8	9.2	10.2	11.8	11.6	11.3	9.8	9.6	11.6
09/10/78	Calm	10.2(P)	P	P	P?	8.4	8.8	8.4	P?	P?	9.0	9.4	9.6	9.8	12.2
09/24/78	WSW	8.4	8.4	9.3	P	9.0	8.0	8.4	8.4	11.2	8.4	8.4	8.4	7.8	8.4
10/08/78	WSW	9.2	8.4	10.8	6.8	7.6	7.6	7.7	9.2	8.4	9.6	9.2	8.6	8.4	8.5
10/22/78	SW	6.8	6.8	8.7	8.0	8.1	7.7	7.2	7.9	7.9	10.0	8.3	8.3	8.4	6.9
11/05/78	SSW	9.6	8.3	P	8.4	8.3	7.9	8.1	7.9	11.6	9.3	8.3	8.3	8.0	P
11/19/78	SW	7.9	7.8	8.3	9.7	7.7	7.5	7.2	7.1	10.2	8.2	8.0	7.7	7.6	7.7
12/03/78	W	7.9	7.9	8.1	8.3	7.2	7.4	9.2	7.5	12.7	7.4	8.5	8.3	7.2	9.2
02/04/79	NW	6.9	7.3	6.2	5.9	5.0	4.7	5.9	6.4	9.2	7.3	6.6	7.1	6.4	7.8
02/18/79	E	6.8	6.5	5.9	6.1	4.5	4.5	4.7	6.3	8.1	6.5	7.9	8.8	8.1	6.5
03/03/79	E	6.3	5.7	5.1	5.5	5.1	5.1	5.5	5.1	8.7	6.9	5.7	5.1	5.1	5.5

P = Plume

Table 15

Radiation Exposure from Airborne Effluent

Location	Aug. 23 - Nov. 15 Net μR/hr	Net mR	Model, mR	Subjective Comparison	Nov. 15 - Mar. 14 Net μR/hr	Net mR
A	1.6	3.2	4.2	G	-0.4	-1.1
B	6.6	13.3	6.0	B	0.6	1.7
C	5.1	10.3	9.0	G	-0.1	-0.3
D	4.1	8.3	5.8	F	1.0	2.8
E	3.0	6.0	5.0	G	0.3	0.8
F	3.0	6.0	2.0	B	-0.4	-1.1
G	----	----	2.7	-	0.6	1.7
H	3.6	7.2	3.9	B	0.1	0.3
I	4.1	8.3	2.8	B	0.9	2.5
J	3.1	6.2	3.0	F	0.6	1.7
K	3.6	7.2	1.9	B	0.4	1.1
L	3.1	6.2	1.6	B	1.4	3.9
M	1.1	2.2	1.6	-	0.1	0.3
N	0.1	0.2	1.7	-	-0.7	-0.2
O	1.1	2.2	2.1	-	0.1	0.3
P	5.1	10.3	6.8	G	0.1	0.3
Average	---				0.3±0.5(1σ)	

Notes: 1. Net μR/hr is TLD average exposure rate (in g/cm² shield) without background. This value was calculated by subtracting from the values in Table 11, 7.3 μR/hr and 6.0 μR/hr during the August 23 - November 15 and November 15 - March 14 periods, respectively, except that at locations E, F, and G, the values subtracted were 1.0 μR/hr less, i.e. 6.3 and 5.0 μR/hr, respectively. These values are the PIC average background exposure rates of 8.9 and 7.3 μR/hr, respectively, divided by 1.22 (see Table 12) to convert to TLD values. The lower background value at locations E, F, and G was similarly adjusted.

2. Exposure periods were 2013 hr during August 23 - November 15, and 2820 during November 15 - March 14.

3. For the model value, monthly exposures were obtained; the exposures were taken to be proportional to the fraction of the month for parts of months.

4. Subjective comparison made only for those data sets where the measured net exposure is greater than 3 mR. G=Good, F=Fair, B=Bad.

III-B. Thermoluminiscent Dosimeter and Gamma Survey Data (continued)

holders E, in mR per period:

Location	Date	TLD	PIC	PIC ÷ 1.22
O	Aug. 2? - Nov. 15	2.2	2.2	1.8
H		7.2	7.9	6.5
O	Nov. 15 - Mar. 14	0.3	0.2	0.2
H		0.3	0.2	0.2

All values are consistent within the uncertainties of measurement.

IV. Conclusion

In this part of the Dresden EDPS, two hypotheses were tested:

(1) that the sector-averaged finite plume model accurately reflected the measured exposure at or near the site boundary, and

(2) that LiF TLDs could accurately measure the station-contributed dose when the current period background was subtracted; the current period background being determined by a survey meter standardized against a PIC-measured exposure rate.

The first hypothesis was found to be incorrect. The sector-averaged model underestimated the measured exposure by approximately 60%, for all practical purposes, a factor of two. Correction factors were determined using two types of multiple sector models (Tables 4 and 5). The most accurate model appears to be the off-axis finite plume model. This model provides realistic and useful estimates of the measured exposures so long as appropriate corrections are made for periods of extremely unstable atmospheric conditions with the monitor bearing close to the downwind wind direction.

The procedure utilizing two PICs and sets of 16 TLDs yields exposure results that permit measurements as low as 6 mR per year, based on a standard deviation of 0.5 μR/hr when the exposure rate from airborne effluent is near 0.1 μR/hr. Determination of differences in the background values at the 14 non-PIC locations should be improved by performing more precise measurements with survey meters and also making PIC measurements at each location, possibly twice each year.

Appendix G

Quality Assurance Program

(1) Current CECo Quality Assurance Articles for Met-orological Monitoring Program. (Rev. 0)

(2) Current Contractor Quality Assurance Program for Meteorological Monitoring Program.

(3) Quality Assurance Articles for Meteorological Monitoring (Rev. 1), Effective 1/1/81.

DATE: 4-2-76

STANDARD QUALITY ASSURANCE ARTICLES

FOR

METEROLOGICAL MONITORING PROGRAM

1.0 QUALITY ASSURANCE SUBMITTALS

1.1 QUALITY ASSURANCE PROGRAM

a. A Contractor who has a controlled copy of his Quality Assurance Program assigned to the Manager of Quality Assurance, Commonwealth Edison Company, need only submit documented verification that the controlled copy is applicable to the Scope of work involved in the bid and include information covering the current effective date of the Program manual including all current revisions in effect.

b. A Contractor who does not have a controlled copy of his Quality Assurance Program assigned to the Purchaser shall submit to the Purchaser, with his bid, three (3) copies of his Quality Assurance Program for review. This program shall meet the requirements of 10CFR50, Appendix B, as described herein.

1.2 QUALITY CONTROL LISTS

a. The Contractor shall include with his proposal an index of the Quality Control procedures he shall employ during the time of contract.

b. The Contractor shall submit with his proposal for inclusion into the contract award, a detailed list of the documents and documentation which will be furnished prior to or concurrent with shipment. This proposal list of documents and documentation shall include all Code required Permanent Records.

2.0 QUALITY ASSURANCE PROGRAM SUBMITTALS AFTER AWARD

2.1 COPIES OF QUALITY ASSURANCE MANUAL

a. After award, the Contractor must submit nine (9*) copies of his accepted Quality Assurance Program to the Purchaser, one of which is a controlled copy assigned to the Manager of Quality Assurance if a Controlled copy has not been previously submitted. If the contract is for more than one station, the Contractor must submit three additional copies for each additional station.

b. After award, the Contractor must submit eight (8) copies of his accepted Quality Assurance Program if the Contractor meets the requirements of 1.1a above.

*Q.A. Department control, P.S.A. (2 copies), Environmental Affairs, General Electric, Purchasing, Station Superintendent, On-site QA, Station Construction

2.2 QUALITY ASSURANCE PROGRAM REVISIONS

a. The control of the accepted Quality Assurance Program is the responsibility of the Contractor. The Contractor shall notify the Purchaser of all revisions to the Quality Assurance Program during the period of his contract. No revisions to the accepted Quality Assurance Program shall be implemented by the Contractor without the Purchaser's acceptance for the work.

3.0 QUALITY ASSURANCE PROGRAM APPROVAL

3.1 APPROVAL REQUIREMENTS

a. Before any Contractor can be considered for an award he must have submitted an acceptable Quality Assurance Program. In order to be considered as acceptable, the program must address the eleven (11) requirements delineated in Article 4.0, "Acceptance Criteria," as applicable.

b. If the Contractor's program does not cover al of the eleven criteria in the required detail, he must state where and how the requirements do not apply.

c. Acceptance by the Purchaser may be based on commitments by the Contractor to revise or amend his Quality Assurance Program to satisfy Purchaser requirements. These commitments shall be made a condition of award.

4.0 ACCEPTANCE CRITERIA

4.1 ORGANIZATION

Contractor's Quality Assurance Program shall include a chart which shows his organization, and the reporting relationship between the Quality Control personnel and management. The Quality Control personnel shall:

a. Have sufficient and well-defined responsibility, authority, qualifications, and organizational freedom to:

   (1) identify quality problems;

   (2) initiate, recommend, or provide solutions, through designated channels;

   (3) verify implementation of solutions; and

   (4) control further processing of a deficiency or unsatisfactory condition until proper dispositioning has occurred.

b. Be independent of operations and scheduling groups.

4.2 QUALITY ASSURANCE PROGRAM

a. The Contractor's Quality Assurance Program must be approved by his management. It shall provide for effective implementation of written policies, procedures, or instructions to ensure that the subject work is accomplished in compliance with the appropriate codes and the project specification.

b. Provisions for controlled distribution and for regular review of the status and adequacy of the total Quality Assurance Program shall be a part of the program.

c. Provisions for training personnel performing activities affecting quality shall be a part of the program.

4.3 INSTRUCTIONS, PROCEDURES AND DRAWINGS

a. Activities affecting quality shall be prescribed by documented instructions, procedures or drawings as appropriate and accomplished in accordance with these documents. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria. Measures to insure the availability, applicability, and accountability of the above shall be described.

4.4 DOCUMENT CONTROL

a. Measures to control issuance of latest applicable documents such as instructions, procedures, drawings and confirmatory documents such as test reports, including changes thereto, which prescribe activities affecting quality shall be described. These measures shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed.

4.5 INSPECTION

a. The program for inspection of activities affecting quality that is established and executed by or for the Contractor and his subcontractors to verify conformance with the documented instructions, procedures, and drawings shall be described. Such inspection shall be performed by individuals other than those who perform the activity being inspected and the results shall be documented.

4.6 EQUIPMENT CALIBRATION

a. A test program shall be established to assure that any testing required to demonstrate that the meterological systems and/or components perform satisfactorily in service is performed by qualified personnel in accordance with written test procedures which incorporate the requirements and acceptance limits contained in the specification. Test procedures shall include provisions for assuring that all prerequisites for the given test have been met; that adequate and calibrated test instrumentation is available and used, and that the test is performed under appropriate environmental conditions. Test results shall be documented and evaluated to assure that test requirements and acceptance limits have been satisfied.

4.7 CONTROL OF MEASURING AND TESTING EQUIPMENT

a. Measures established to assure that tools, gages, instruments and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated and adjusted at specified periods to maintain accuracy within limits shall be described.

4.8 INSPECTION AND TEST STATUS

a. Measures established to indicate, by the use of markings, such as stamps, labels, routing cards, etc., the status of inspections and tests performed upon individual items being furnished under this specification shall be described.

4.9 CORRECTIVE ACTION

a. Measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. The identification of the adverse condition, the cause of the condition, and the corrective action taken to prevent future occurrence of like deficiencies shall be documented and reported to appropriate levels of management.

4.10 QUALITY ASSURANCE RECORDS

a. The records that will be maintained to furnish evidence of activities affecting quality and the measures that will ensure prompt and complete delivery to Purchaser of those documents required by this specification shall be described. The records shall be identifiable and retrievable. The Contractor shall meet the requirements of all applicable codes concerning record retention, such as duration, location, and assigned responsibility. The Contractor shall notify the Purchaser prior to disposing of records which have been retained by him for the duration specified in his Quality Assurance Program.

4.11 AUDITS

a. A comprehensive program of planned and periodic audits to be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program shall be described. The audits shall be performed in accordance with the written procedures or checklists by appopirately trained personnel not having direct responsibilities in the areas being audited. Audit results shall be documented and reviewed by management having responsibility in the areas audited. Follow-up action, including reaudit of deficient areas, shall be taken when indicated.

5.0 QUALITY CONTROL PROCEDURES SUBMITTALS

a. Within twelve (12) weeks after award of Contract, the Contractor should submit the detailed procedures to be used. Procedures governing work that is to be performed shall be submitted to the Purchaser for review and acceptance.

NOTE: The Contractor shall not start any work covered by these procedures until the appropriate procedure has been accepted in writing by the Purchaser.

b. The Quality Control procedures shall contain those administrative procedures necessary to implement each section of this Quality Assurance Plan. The procedures shall designate who is responsible for the implementation of each of the requirements stated in the Quality Assurance Plan and define the authority and duties of all personnel associated with Quality Control. The procedures shall detail how all elements affecting the product quality will be processed and shall include the specification of the necessary documentation.

EXAMPLES: Internal audit, document control, etc.

6.0 INSPECTION POINT PROGRAM SUBMITTAL

The Purchaser and/or his designated representative shall have full access to Contractor's and Subcontractor's shops for reviewing progress and determining acceptability of Quality Control Work.

The Station Construction Site Project Superintendent or Superintendent of an Operating Station, as applicable, or his designee shall be notified at least three (3) working days, (excluding Saturdays and Sundays), prior to start of specified field tests and calibrations.

7.0 DOCUMENTATION SUBMITTAL

All documentation shall be clear, legible and of suitable quality for microfilming and/or storage for the life of the plant.

8.0 SPARE PARTS

All requirements regarding quality control and documentation that apply to the original parts of the specified equipment shall apply equally to the spare parts of the specified equipment.

5 December 1979
Revision 2

Quality Assurance Program

for

Meteorological Monitoring Programs

Prepared by

MURRAY AND TRETTEL, INCORPORATED
Northfield, IL 60093

26 July 1976

Controlled Distribution No. 9

TABLE OF CONTENTS

Foreword

This is Murray and Trettel's Quality Assurance Program which describes the requirements that must be implemented in connection with the Commonwealth Edison Company meteorological monitoring programs.

The report is divided into eleven (11) sections conforming in format to eleven (11) criteria specified in Appendix Three to the "Specifications For 1977-78 Meteorological Monitoring Service and Maintenance", April 1976, Revision 2.

The contents of this report are to be considered as Murray and Trettel policy and, as such, are to be followed by all employees to the extent of their involvement in the monitoring program.

John R. Murray

John R. Murray, B.S., J.D.
Certified Consulting Meteorologist
President

22 July 1976

Introduction

This report has been prepared to delineate the requirements governing the Murray and Trettel Quality Assurance Program for meteorological monitoring programs. Implementation of the monitoring program with detailed procedures provides the degree of quality assurance commensurate with the requirements of applicable codes and requirements of agencies which govern the installation and operation of meteorological monitoring equipment, and the handling, reduction and processing of data. The scope of this report covers the total Quality Assurance Program for the life of the monitoring program.

1. Organization

Murray and Trettel, Incorporated is responsible for the assurance of quality in all phases of the acquisition, reduction, and analysis of meteorological monitoring data. Murray and Trettel executes this responsibility in accordance with the program described herein and assigns areas of ultimate responsibility to specific individuals.

Lines of authority and responsibility for the Quality Assurance Program are documented in the form of an organization chart. Key quality assurance positions including those providing technical support or audit responsibility are described. The organization chart for the meteorolo ical monitoring program is shown in Figure 1. Solid lines represent responsibility for implementing the procedures and instructions. Dashed lines represent audit responsibility for verifying compliance with the procedures and instructions. The Quality Assurance Office acts independently of the person or group directly responsible for performing the activites of the meteorological program.

The specific responsibility for the Quality Assurance Program are described in the following paragraphs.

Executive Vice President

The Executive Vice President of Murray and Trettel has the overall responsibility for the Quality Assurance (QA) of the meteorological monitoring programs. The development of quality assurance policy for environmental studies is under his jurisdiction.

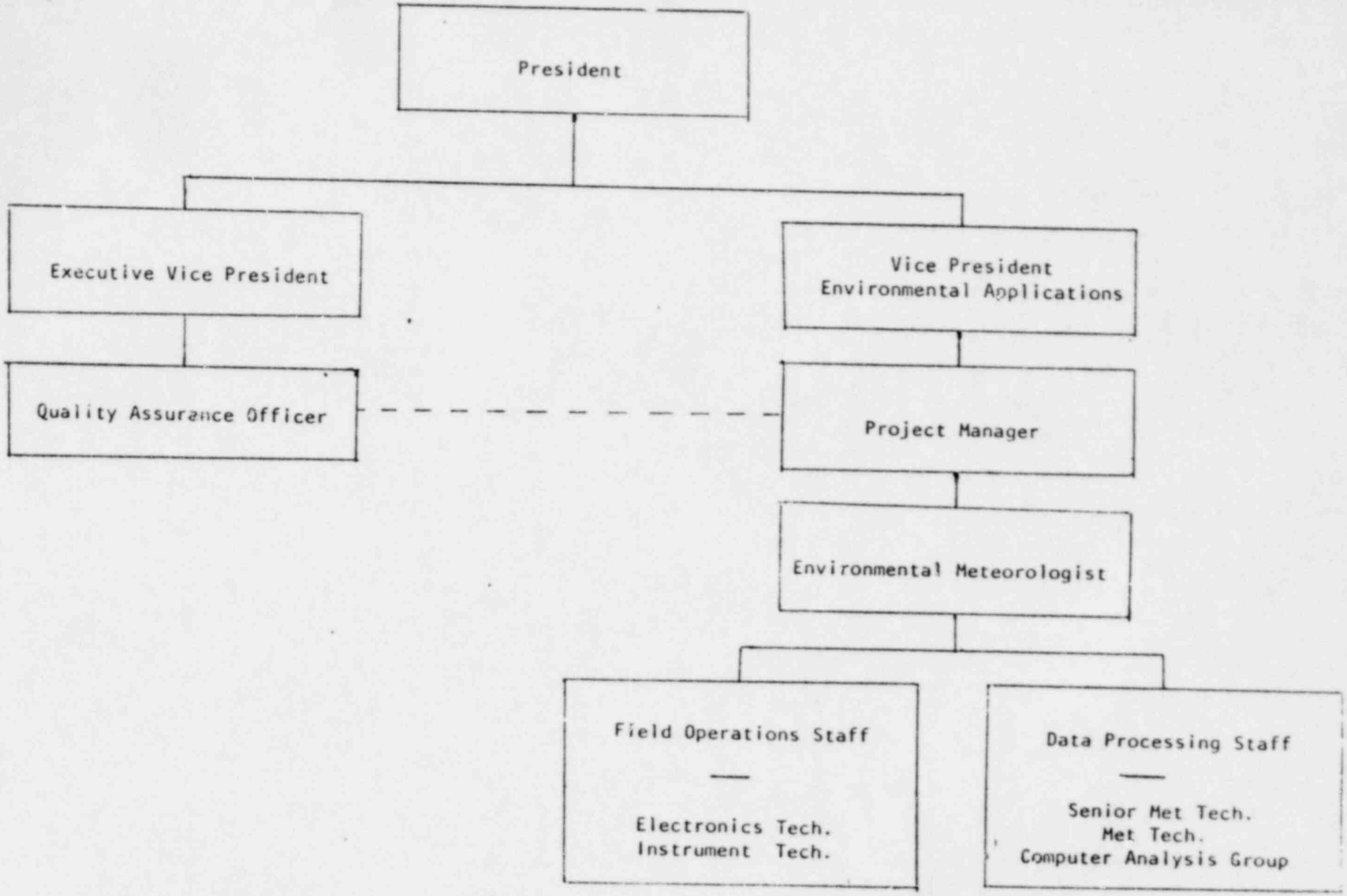


Figure 1: Murray and Trettel, Incorporated Meteorological Monitoring Programs Organization.

Quality Assurance Officer

Authority and responsibility to conduct periodic audits is assigned by the Executive Vice President to the Quality Assurance Officer. He reports directly to the Exectuive Vice President in all matters involving quality assurance, and is independent of the normal operation of the meteorological monitoring programs except for matters involving quality assurance.

The Quality Assurance Officer is responsible for conducting audits and inspections, detecting deficiencies in the procedures, and recommending improvements in the procedures if deficiencies are discovered.

Vice President, Environmental Applications

The Vice President, Environmental Applications is responsible for all environmental projects, including the meteorological monitoring programs. The Vice President, Environmental Applications is also responsible for the training of personnel involved in the program, and for approving all procedures and manuals used in the program.

Project Manager

The Project Manager has the overall direct responsibility for the monitoring program. He is responsible for providing technical assistance, assigning time tables, setting priorities and the day-to-day decisions required by the project. He is also responsible for the preparation of procedures to assure data validity.

Environmental Meteorologist

The Environmental Meteorologist is responsible for the day-to-day operation of the project, and for providing technical assistance and training to the technicians. The duties also include inspection of charts and records of project documents for completeness, final editing of the data record and preparation of monthly, semi-annual and other miscellaneous related reports.

Data Processing Staff

The Data Processing Staff maintains project records, reviews strip charts, reduces data and performs other tasks related to the day-to-day operation of the program.

Field Operations Staff (Field Staff)

The Field Staff maintains the field equipment, performs in-situ/instrument calibrations, provides documentation of the performance of each system, maintains a spare parts inventory and maintains service instrumentation in proper calibration.

2. Quality Assurance Program

The Quality Assurance Program at Murray and Trettel is approved by management and assures effective implementation of program procedures. These procedures or instructions assure the monitoring program is conducted in compliance with the appropriate codes and program specifications. In general, the Quality Assurance Program verifies (through audits) that activities have been correctly performed. Audit personnel are independent of the activities being audited. Quality Assurance personnel have sufficient authority and organizational freedom to identify problems, to initiate, recommend or provide solutions, and to verify implementation of the solutions. Any disagreements on procedures are resolved through a review of the situation by the president of Murray and Trettel.

Regular reviews of the status of adequacy of the monitoring program are provided through a series of inspections and audits conducted by the QA personnel. A controlled distribution list is set up and recepients of controlled documents will recieve any alterations or revisions. Training is provided to all new personnel, and to all personnel when new procedures are incorporated into the program.

Instruments are maintained by qualified personnel and the equipment used for the calibration of the meteorological systems are themselves calibrated on a routine basis.

3. Instructions, Procedures and Drawings

A set of procedures for meteorological monitoring programs has been prepared for use by all personnel involved with the program [1]. These procedures contain instructions, specifications, and check lists that cover all phases of the monitoring program from the sensing of the meteorological data to its final verification, analysis and storage.

The procedures manual is maintained by the Project Manager. All persons having registered copies of the manual receive revisions as they are approved and implemented.

All revisions to the procedures manual are approved by CECO before being implemented by Murray and Trettel per sonnel.

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1 Meteorological Monitoring Program: Equipment Servicing and Data Recovery Procedures. A controlled document No. 1084 dated 7/79.

4. Document Control

A document control system is used to assure that documents such as procedures, specifications, maintenance forms and data handling forms are reviewed for accurracy and approved by authorized personnel. Such documents are distributed to and used by the personnel responsible for their use. Changes to these documents are handled similarly and are reviewed and approved by the same personnel that performed the original review and approval.

A master controlled distribution list is used to designate the recipients of the documents. Each document recipient is responsible for insuring that only the latest authorized procedures are in use and void documents are so identified.

The Project Manager is responsible for instituting the document control system for the project and the Environmental Meteorologist is responsible for assuring the necessary files, logs, and procedures are instituted and maintained in a neat and proper manner. The Environmental Meteorologist is also responsible for assuring that those documents that are to be sent to the client are prepared and transmitted in a timely manner.

Documents pertaining to the maintenance, calibration, and performance of equipment are retained in a central filing system at Murray and Trettel.

5. Inspection

Inspections are carried out at all stages of the data acquisition, processing, and reporting to assure that all procedures are being followed in the correct manner. The inspections are performed by personnel other than those who perform the task.

Documents that have been inspected are initialed or signed by the individual inspecting the document. An incomplete form is returned to the individual responsible for it. Corrections are made before being initialed by the inspector.

Documents to be inspected are:

1) Weekly Visitation Logs
2) Bi-Monthly Routine Maintenance Forms
3) Emergency (Non-Routine) Maintenance Forms
4) Batches In-House
5) Batch Condition Sheets
6) Data Recovery and Equipment Status Report
7) Digitizer Tape Listing
8) Tape Explanation Sheet
9) Digitizing Record Sheet
10) Monthly Reports
11) Semi-Annual Reports

6. Equipment Calibration

A testing program has been established to assure that the meteorological sensors, signal conditioners, and recorders are performing in the required manner. Calibrations are conducted at specified intervals by trained personnel. In addition, calibration of the equipment is performed whenever it has been repaired or whenever the quality assurance checks, made on the data, indicate that the system may not be performing up to specifications.

The test equipment used by Murray and Trettel field personnel is calibrated at routine intervals. Electronic test equipment is calibrated and certified by the manufacturer and thermometers are calibrated in house by qualified technicians.

To assure that all of the required tests and calibrations are performed, specific forms have been developed for each site. These forms serve to remind the technician of the required tests, to document the results of the tests and to indicate any problems encountered in the procedure. The acceptable tolerances for each test are provided on the form to assure all calibrations are within acceptable limits. The calibrated systems are affixed with a sticker indicating the date of calibration, the initials of the technician who performed the work, and the date of the forthcoming calibration.

7. Control of Measuring and Testing Equipment

The elctronic instruments and thermometers used to calibrate the meteorological systems are themselves calibrated at routine intervals. This assures that these items are maintained within acceptable limits of accuracy.

Electronic instrumentation is calibrated once each year by the manufacturer in such a manner that the results can be traced to the National Bureau of Standards. These results are certified by the manufacturer and a calibration label is affixed to the instrument. The label states the date of the calibration and date the next calibration is due.

Thermometers are calibrated at Murray and Trettel by trained personnel. A water bath and a precision thermometer, whose calibration is traceable to the National Bureau of Standards, are used in the calibration procedure. An eight point calibration is performed quarterly on each thermometer and the results are documented. New thermometers are calibrated before use in the field.

8. Inspection and Test Results

The status of inspection and tests performed on items furnished as part of the program is indicated by means of labels affixed to the items. All instrumentation used in the calibration of the meteorological system have calibration labels indicating when they were last calibrated, and the date their next calibration is due.

Each time the system is calibrated, a calibration label is affixed to the system by the field service personnel. This label indicates the date of the calibration, the personnel who performed the calibration, and the date the next calibration is due.

9. Corrective Action

A series of on-site visits, checks, and a weekly review of the strip charts provide several means of detecting problems at an early stage.

Procedures have been developed to identify promptly any problems in the data base. When defective equipment is identified as the problem source, field service personnel are notified and a site visit is scheduled to correct the problem. It is not possible to eliminate all data loss from the meteorological systems, but it is possible to minimize the loss through quick detection of the problem. The cause of each problem is identified and documented in the routine course of the project. When applicable, recommendations of modifications to instrumentation or procedures are made in order to eliminate or minimize the loss of data.

10. Quality Assurance Records

Records are maintained to furnish evidence of activities affecting the quality of the data collected by the monitoring programs. All records of site visits, routine maintenance, exceptional maintenance, data review, and progress reports are retained as part of the quality assurance program.

The timely submission of the reports required by the program specification is assured by the routine inspection and processing of the data within one month of the date collected.

All quality assurance records are identifiable and retrievable. Notification will be given to CECo prior to disposing of these records and disposal will not be allowed until permission from CECo is obtained. All quality assurance records are maintained in accordance with applicable codes regarding record retention.

11. Audits

Audits are performed by Murray and Trettel personnel who are not involved in the day-to-day operation and management of the project. The audits verify the implementation and effectiveness of the monitoring program.

Audits cover maintenance and calibration of tower acquisition systems, data handling, and data reduction. Procedures and inspections of records are included in the audit.

The audits are conducted twice each year by the Quality Assurance Officer using checklists or an agenda approved by the Executive Vice President.

A report is written after each audit and consists of the following: a Summary Sheet, Checklists or agenda, and any additional pertinent details recorded on additional sheets necessary to support the findings. This report is submitted to the Executive Vice President for review and is retained as a part of the quality assurance documentation.

A follow-up review of deficient areas or adverse conditions and on corrective action commitments, is carried out to assure effective implementation.

Deficiencies in the execution or implementation of corrective action are brought to the attention of the person responsible for their rectification. Continued deficiencies or failure to implement corrective action are reported, in writing, to appropriate executives within Murray and Trettel.

Commonwealth Edison Company

Quality Articles for Meteorological Monitoring

Section I - Quality Assurance

1.0 Quality Assurance Program:

1.1 The contractor shall be required to have an acceptable Quality Assurance Program which will be in effect for the duration of the contract. The Quality Assurance Program shall include the quality assurance system, organization, policies, responsibilities, listing of procedures and/or requirements for processes necessary to control quality throughout all phases of the contract. This program shall meet the applicable requirements of 10CFR50, Appendix B and must address the requirements delineated in Article 5.0, "Acceptance Criteria". Acceptable guides for meeting the applicable requirements are ANSI N 45.2, "Quality Assurance Requirements for Nuclear Power Plants", and applicable associated ANSI N 45.2 daughter documents. The program must be accepted by the purchaser prior to award of contract.

2.0 Quality Assurance Program Approval:

2.1 Before any contractor can be considered acceptable for an award of contract, he must have submitted an acceptable Quality Assurance Program. In order to be considered as acceptable, the program must address, as applicable, the requirements delineated in Article 1.0 above.

2.2 If the contractor's program does not cover all of the requirements in detail, he must state when and how the requirements do not apply. This statement of non-applicability must be substantiated.

2.3 Commitments accepted by the purchaser as a condition of award shall be implemented by the contractor immediately upon award of contract. These commitments shall require the contractor to make written changes to the program in the form of revisions or supplements to the program. The supplement shall be controlled in the same manner as the manual, and considered as a auditable part of the program.

2.4 The control of the accepted Quality Assurance Program is the responsibility of the contractor. Contractor shall promptly notify the purchaser of all revisions to the Quality Assurance Program for the duration of the contract. No revisions to the accepted Quality Assurance Program shall be implemented on the purchaser's work by the contractor without the purchaser's written approval of the Program revision.

3.0 Quality Assurance Program Submittal with Proposal:

3.1 A contractor who has written acceptance by the purchaser and a controlled copy of the accepted Quality Assurance Program assigned to the Manager of Quality Assurance, Commonwealth Edison Company, need only submit documented verification that the controlled copy is applicable to the scope of work involved in the bid, and include information with the proposal covering the current effective date of the program manual, including all current revisions and supplements in effect.

3.2 A contractor who does not have an accepted and controlled copy of his Quality Assurance Program as described in Article 3.1 above shall submit to the purchaser with his bid two (2) controlled copies of his Quality Assurance Program for review and acceptance, one assigned to the Manager of Quality Assurance and the other assigned to the Nuclear Stations Division Manager.

4.0 Quality Assurance Program Submittal After Award:

4.1 After award, if the contractor meets the requirements of 3.2 above, he must submit three (3) uncontrolled copies of the accepted Quality Assurance Program to the purchaser.

4.2 After award, if the contractor meets the requirements of 3.1 above, the contractor must submit four (4) copies of the accepted Quality Assurance Program. One (1) copy must be controlled, and will be assigned to a designated individual in the Nuclear Stations Division; the three (3) remaining shall be uncontrolled.

4.3 After award, if contract is for more than one station, the contractor must submit two (2) additional uncontrolled copies for each additional station.

4.4 After award, any revisions to the accepted quality assurance program which the purchaser approves, the contractor must submit copies of the accepted revisions for the uncontrolled and controlled manuals in the purchaser's possession for the duration of the contract.

5.0 Quality Assurance Program Acceptance Criteria:

5.1 Organization:

A. The contractor's Quality Assurance Program shall include an organization chart identifying key positions and the reporting relationship between the Quality personnel and management (including field Q.A. organization, if applicable). All quality related activities which are referenced in the manual must be assigned to specific personnel. The Quality Assurance personnel shall have:

1. Written responsibilities for quality related job positions.
2. Authority and organizational freedom to:
    a. identify and evaluate problems
    b. require and implement approved corrective actions
    c. control further activities where appropriate action such as "stop work" may be required.
3. Independence from groups involved in design and/or operation of the system, computer programming, data processing system design/modification.

5.2 Quality Assurance Program:

A. The contractor's Quality Assurance Program must be formally accepted by Company Management with a written policy statement. This Program shall be implemented through written procedures and/or instructions or they shall be established to ensure that the subject's work is accomplished in compliance with the appropriate code and procurement requirements.

B. Provisions for training Quality Assurance personnel performing activities affecting quality shall be a part of the program. These provisions must include how this training is accomplished and who is responsible for its implementation.

C. Provisions for a review of the status, adequacy, implementation and effectiveness of the total Quality Assurance Program on a specific time schedule shall be a part of the Program.

D. Provisions shall be established in the Program for the controlled issuance of the latest revision to the quality assurance manual, procedures and instructions.

E. Includedin the Program is a commitment that the program complies with applicable portions of 10 CFR 50 Appendix B and/or ANSI N 45.2.

F. The Program shall delegate responsible individual(s) to sign off on Certificates of Conformance and/or

Compliance

5.3 Design Control:

A. Measures to assure that the design basis for the systems and/or components are correctly translated into specifications, drawings, procedures, and instructions as appropriate, shall be described. These measures shall include provisions to insure that appropriate quality standards are specified and included in design documents.

B. The design control measures for independent verification or check of the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program shall be described.

C. Means by which the contractor will insure that design changes are subjected to design control measures commensurate with those applied to the original design shall be described.

5.4 Procurement Document Control:

A. Measures to assure that purchase documents for procurement of material, equipment, and services, whether purchased by the contractor or by a subcontractor performing a significant portion of the actual services, are reviewed for inclusion of quality requirements shall be described in the Program.

Subcontractors who perform a significant portion of the service shall be required to provide to the contractor a Quality Assurance Program consistant with the requirements of the contractor's Q.A. program for review and acceptance by the contractor. The contractor will be responsible for determining the Quality Assurance requirements to be applied to any subcontractor who performs a significant portion of the actual services.

5.5 Instructions, Procedures and Drawings:

A. Activities affecting quality shall be prescribed by documented work procedures or instructions, as appropriate, and accomplished in accordance with these documents. Procedures or instructions shall include appropriate acceptance criteria for work performance and quality compliance. The above measures shall be described in the Program.

5.6 Document Control:

A. Measures to control the issuance of the latest applicable documents such as instructions, procedures, drawings, purchase requirements and confirmatory documents such as test reports, including changes thereto, which prescribe activities affecting quality shall be described. These measures shall assure that documents including changes are reviewed for adequacy and approval for release by authorized personnel and are distributed to and used at locations where the prescribed activity is performed; and shall assure that obsolete drawings, specifications and instructions have been destroyed or isolated from use.

5.7 Control of Purchased Material, Equipment and Services:

A. Measures to assure that purchased material, equipment and services, whether purchased directly or through subcontractors, conform to the procurement documents shall be described. These measures shall include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by contractor or subcontractor, inspection at the contractor or subcontractor source, and receiving inspection for compliance with procurement documents upon delivery. The effectiveness of Quality Control by contractor, or by subcontractors who perform a significant portion of the actual services, may be assessed by purchaser or his designee at intervals appropriate to the importance, complexity, and quantity of the activities being performed.

5.8 Inspections:

A. The inspection program for activities affecting quality that is established and executed by or for the contractor and his subcontractors to verify conformance with documented instructions procedures, and drawings shall be described. Such inspection shall be performed by qualified personnel, with certification as required, other than those who perform the activities being inspected. The program shall identify the person responsible for the training, documentation of this training, and maintenance of the training records.

B. There shall be provisions in the Program for establishing, after award of contract, inspection, by the customer or by other as directed by Edison, of any activities or facilities utilized in the performance of these services by the contractor or significant subcontractors.

C. The Program shall have provisions for documenting and retaining all inspection results:

5.9 Test Control:

A. A test program shall be established to insure that any bench or field testing required to demonstrate that the systems and/or components perform satisfactorily in service is performed by qualified personnel, with certification as required, in accordance with written test procedures which incorporate the requirement and acceptance criteria and limits contained in specifications. Test procedures shall include provisions for assuring that all prerequisites for the given test have been met, that adequate and calibrated test instrumentation is available and used, and that the test is performed under appropriate environmental conditions.

B. The Program shall have provisions for documenting, evaluating, and retaining all test results.

5.10 Control of Measuring and Test Equipment:

A. Measures shall be established in the Program to assure that proper tools, gauges, instruments and other measuring and testing devices are used in activities affecting quality. To assure accuracy the equipment shall be calibrated, adjusted and maintained at prescribed intervals or prior to use against certified equipment having known valid relationships to the National Bureau of Standards or other recognized applicable standards.

B. Records shall be maintained and equipment suitably marked (such as tag, sticker, etching, etc.) to verify calibration status.

5.11 Handling, Storage and Shipping:

A. Measures established to protect equipment being transported or in storage against damage or deterioration shall be described.

5.12 Nonconforming Materials, Parts or Components:

A. Measures established to control materials, parts or components which do not conform to requirements in order to prevent their inadvertent use or installation shall be described. These measures shall include procedures for identification, documentation, segregation and disposition.

5.13 Corrective Action:

A. Measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. The identification of the adverse conditions, the cause of such condition and the corrective action taken to prevent continuing recurrence of like conditions shall be documented and reported to appropriate levels of management and the customer.

5.14 Quality Assurance Records:

A. Records shall be maintained to furnish evidence of activities affecting Quality. The contractor shall establish measures that will assure prompt and complete delivery to the purchaser of any documents required by the specification. The contractor shall meet the requirements of applicable codes and ANSI Standards concerning record retention regarding identifiability and retrievability, duration of retention, location, and assigned responsibilities.

5.15 Audit:

A. Measures established to provide a comprehensive program of planned and scheduled audits to be carried out to verify compliance with all aspects of the Program, and to determine the effectiveness of that Program, shall be described. This plan shall include both scheduled internal audits and, where appropriate, audits of subcontractors who perform a significant portion of actual services.

B. The Program shall provide for audits to be conducted in accordance with written procedures and/or checklists by trained and certified audit personnel not having direct responsibilities in areas being audited. A description shall be provided in the Program of Auditor training activities, with qualification and certification requirements. This shall include a description of training activities, a delegation of responsibilities for performance of these activities, and documentation of these activities.

C. Audit results shall be documented with objective evidence, distributed, and an archival file shall be maintained. The audit results shall be reviewed by management having responsibilities for the area being audited.

D. Follow-up action, including re-audit of the deficient areas(s) to assure corrective action has been accomplished, shall be described in Program.

Section II - Quality Control

1.0 Quality Control Document Submittal with Proposal

A. The contractor shall include with his proposal an index of Quality Control Procedures to be applied to the work.

B. The contractor shall submit with his proposal for inclusion into the contract awarded, a detailed list of the quality records and documentation regarding system operations and activities, other than those required by the specification, which will be furnished to, or available for inspection by the purchaser.

2.0 Quality Control Document Submittal after Award:

2.1 Quality Control Procedures:

A. Within six weeks after an award of contract, the contractor should submit the detailed procedures to be used or a schedule for submitting these procedures.

NOTE: A contractor shall not start any work covered by these procedures until the appropriate procedures have been accepted in writing by the purchaser and/or the purchaser's consulting engineer as appropriate.

B. The Quality Control Procedures shall contain those administrative procedures necessary to implement each Section (5.1 through 5.15) of the Quality Assurance Program described above. The procedures shall designate who is responsible for the implementation by each of the departments stated in the Quality Assurance Program and define the authorities and duties of all personnel associated with quality control. The procedures shall detail how all elements affecting the product quality will be processed and shall include the specification of the necessary documentation.

C. Quality Control Procedures shall also contain those design, testing, inspection, cleaning, etc., procedures necessary for the accomplishment of the work and to assure its proper quality. Procedures shall be qualified as necessary to Code or Standard requirements. These procedures shall detail what equipment is to be used, limiting conditions, acceptance criteria, techniques, etc., that will be used.

2.2 Inspection Program:

A. An inspection program shall be established by the contractor and shall include pertinent maintenance and inspection operations which will be of concern to the purchaser relative to Quality Control. Contractor's recommend calibration and maintenance program will be applied to the equipment, and documented in monthly reports. Inspection programs shall be submitted to Nuclear Stations Division Manager or his designee.

NOTE: 1. The contractor shall not start any work which requires an inspection program until the purchaser or the purchaser's consulting engineer has reviewed and accepted the program as appropriate.

B. Purchaser and/or his designated representative shall have full access to contractor's and subcontractor's shops and field sites for reviewing progress and determining acceptability of Quality Control activities. Nuclear Stations Division Manager or his designee shall be notified at least two (2) working days, excluding Saturdays and Sundays, prior to starting of specified installation, calibration, or test programs.

C. Purchaser and/or his designated representative shall have full access to contractor's and subcontractor's shops for reviewing and auditing the implementation of its quality assurance program. Any findings resulting from a contractor's/subcontractor's audit shall be addressed and promptly corrected to the purchaser's satisfaction. The audited organization shall provide a scheduled date for completion of corrective action.

2.3 Subcontractor Requirements:

A. Contractor shall be responsible for the review, comment and acceptance of the Quality Assurance Program and Quality Control Procedures of the subcontractor who performs a significant portion of actual services. In addition, contractor shall be responsible for the subcontractor's work.

2.4 Nonconformance Report:

A. Any nonconformance with purchase documents, approved drawings, procedures, or approved material selection shall be promptly reported in writing to the Purchaser.

2.5 Quality Control Records:

A. Copies of all appropriate documentation as herein specified or as required by applicable Codes, Standards, or criteria, shall be submitted in monthly and semi-annual reports.

2.6 Certificate of Compliance/Conformance:

A. Certificate of Compliance

A Certificate of Compliance signed by a qualified party, attesting that the items or services are in accordance with the customer's purchase order and specification, and accompanied by all documentation required by these articles to substantiate that statement, is required upon commencement of the services contemplated by this contract.

2.7 Invoice Submittal:

A. Invoices for equipment purchased for customer shall be sent to Nuclear Stations Division, Commonwealth Edison Company

2.8 Spare Parts:

A. All requirements regarding Quality Control and documentation that applied to the original parts of the specified equipment shall apply equally to the spare parts of the specified equipment. Contractor shall identify those requirements in detail on spare parts quote.

ITEM F. 3 OF ZION CONFIRMATORY ORDER,

DATED FEBRUARY 29, 1980

APPLICABLE PARTS OF TITLE 10

COMMONWEALTH EDISON COMPANY

ZION STATION, UNITS 1 AND 2

AUGUST 1980

INTRODUCTION

This document is a report of the conformance of Zion Station design and operation to 10CFR20 and 10CFR50. This report complies with Item F.3 of Appendix A to the NRC Confirmatory Order for Zion Station dated February 29, 1980, and letter from H. R. Denton (NRC) to D. L. People (CECo) dated February 29, 1980.

ITEM F.3 OF
ZION CONFIRMATORY ORDER, DATED FEBRUARY 29, 1980
APPLICABLE PARTS OF TITLE 10

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10 CFR 20.101 - RADIATION DOSE STANDARDS FOR INDIVIDUALS IN RESTRICTED AREAS

STATEMENT OF SECTION 20.101 - PARAGRAPH (a)

In accordance with the provisions of Section 20.102 - Paragraph (a), and except as provided in Paragraph (b) of this section, no licensee shall possess, use, or transfer licensed material in such a manner as to cause any individual in a restricted area to receive in any period of one calendar quarter from radioactive material and other sources of radiation a total occupational dose in excess of the standards specified in the following table:

	Rems per Calendar Quarter
1. Whole body; head and trunk; active blood-forming organs; lens of eyes; or gonads..............................	1-1/4
2. Hands and forearms; feet and ankles...	18-3/4
3. Skin of Whole body...................	7-1/2

EVALUATION OF COMPLIANCE

The personnel occupational dose exposure limits at Zion are the same as listed above (Zion Radiation Protection General Procedures, RP-1190-1).

STATEMENT OF SECTION 20.101 - PARAGRAPH (b)

A licensee may permit an individual in a restricted area to receive a total occupational dose to the whole body greater than that permitted under paragraph (a) of this section, provided:

(1) During any calendar quarter the total occupational dose to the whole body shall not exceed 3 rems; and

(2) The dose to the whole body, when added to the accumulated occupational dose to the whole body, shall not exceed 5 (N-18) rems where "N" equals the individual's age in years at his last birthday; and

(3) The licensee has determined the individual's accumulated occupational dose to the whole body on Form NRC-4, or on a clear and legible record containing all the information required in that form; and has otherwise complied with the requirements of Section 20.102. As used in paragraph (b), "Dose to the whole body" shall be deemed to include any dose to the whole body, gonads, active blood-forming organs, head and trunk, or lens of eye.

EVALUATION OF COMPLIANCE

1. Zion Radiation Protection General Procedures, RP-1190-1

Personnel doses at Zion shall be controlled as follows:

a. Each individual must have supervisory approval before exceeding a daily whole body dose of 50 mrem.

b. Daily whole body doses in excess of 100 mrem or weekly doses in excess of 300 mrem must be approved by radiation protection supervision.

c. Quarterly whole body doses in excess of 1250 mrem can be approved by the assistant superintendent after discussion between management and bargaining group representatives to a maximum of 3000 mrem.

d. Annual whole body doses in excess of 5000 mrem can be approved by the station superintendent after discussions between management and bargaining group representatives. Annual whole body doses will not exceed 7000 mrem except by mutual agreement between management and bargaining group representatives.

2. Determination of Personnel Radiation Exposure, RP-1210-1

All personnel who have been issued a film badge at Zion Station will have either an active or inactive Form NRC-4 file; this Form 4 is the base file or mechanism for establishing a computer-based occupational exposure history main file (Form NRC-5).

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.101.

REFERENCES

Zion Procedures

RP-1190-1 Zion Radiation Protection General Procedures

RP-1210-1 Determination of Personnel Radiation Exposure

10 CFR 20.102 - DETERMINATION OF PRIOR DOSE

STATEMENT OF SECTION 20.102 - PARAGRAPH (a)

Each licensee shall require any individual, prior to first entry of the individual into the licensee's restricted area during each employment or work assignment under such circumstances that the individual will receive or is likely to receive in any period of one calendar quarter an occupational dose in excess of 25 percent of the applicable standards specified in Section 20.101 - Paragraph (a) and Section 20.104 - Paragraph (a), to disclose in a written, signed statement, either (1) that the individual had no prior occupational dose during the current calendar quarter, or (2) the nature and amount of any occupational dose which the individual may have received during that specifically identified current calendar quarter from sources of radiation possessed or controlled by other persons. Each licensee shall maintain records of such statements until the Commission authorizes their disposition.

EVALUATION OF COMPLIANCE

1. Film Badge Program, RP-1210-2

All persons receiving a film badge will completely prepare Form NRC-4 prior to being issued a film badge. The Form NRC-4 contains the information required in 10 CFR 20.102 - Paragraph (a). A badge will not be issued if the form is not complete.

2. Determination of Personnel Radiation Exposure, RP-1210-1

All personnel who have been issued a film badge at Zion Station will have either an active or inactive Form NRC-4 file.

3. FSAR, Section 12.3.3.3

A film badge is issued to all personnel who are permanently assigned to Zion Station.

4. FSAR, Section 12.3.3.4

Personnel monitoring equipment (film badge) is issued to all personnel entering a controlled area under such circumstances that they are likely to receive a dose in any calendar quarter in excess of 25 percent of the applicable limit Procedure RP-1210-2, Film Badge Program).

STATEMENT OF SECTION 20.102 - PARAGRAPH (b)

Before permitting, pursuant to Section 20.101 -Paragraph (b), any individual in a restricted area to receive an occupational radiation dose in excess of the standards specified in Section 20.101 - Paragraph (a), each licensee shall:

(1) Obtain a certificate on Form NRC-4, or on a clear and legible record containing all information required in that form, signed by the individual showing each period of time after the individual attained the age of 18 in which the individual received an occupational dose of radiation; and

(2) Calculate on Form NRC-4 in accordance with the instructions appearing therein, or on a clear and legible record containing all the information required in that form, the previously accumulated occupational dose received by the individual and the additional dose allowed for that individual under Section 20.101 - Paragraph (b).

EVALUATION OF COMPLIANCE

All persons receiving a film badge will completely prepare Form NRC-4 prior to being issued a film badge. A badge will not be issued if this form is not completed (Film Badge Program, RP-1210-2).

Form NRC-4 will be filled out in accordance with instructions printed on the reverse side of the form, particularly:

a. If an individual has an exposure history, he is to attach it to the form.

b. If an individual does not have his exposure history, an appropriate request for exposure history must be completed.

c. If an individual has never been occupationally exposed to radiation, he is to enter none.

All Form NRC-4's must be signed and dated in order to be valid (Film Badge Program, RP-1210-2).

STATEMENT OF SECTION 20.102 - PARAGRAPH (c)

(1) In the preparation of Form NRC-4, or on a clear and legible record containing all the information required in that form, the licensee shall make a reasonable effort to obtain reports of the individual's previously accumulated occupational dose. For each period for which the licensee obtains such reports, the licensee shall use the dose shown in the report in preparing the form. In any case where a licensee is unable to obtain reports of the individual's occupational dose for a previous complete calendar quarter, it shall be assumed that the individual has received the occupational dose specified in whichever of the following columns apply:

Part of Body	Column 1 Assumed exposure in rems for calendar quarters prior to Jan. 1, 1961	Column 2 Assumed exposure in rems for calendar quarters beginning on Jan. 1, 1961
Whole body, gonads, active blood-forming organs, head and trunk, lens of eye.	3 3/4	1 1/4

(2) The licensee shall retain and preserve records used in preparing Form NRC-4 until the Commission authorizes their disposition.

If calculation of the individual's accumulated occupational dose for all periods prior to January 1, 1961 yields a result higher than the applicable accumulated dose value for the individual as of that date, as specified in paragraph (b) of Section 20.101, the excess may be disregarded.

EVALUATION OF COMPLIANCE

1. Film Badge Program, RP-1210-2

If an individual does not have his exposure history, an appropriate request for exposure history must be completed. Until the exposure history (ies) are received for an individual, it is necessary to enter 1250 mrem/quarter for each quarter of work in a radiation facility since January 1, 1961. If exposure was received prior to January 1, 1961, 10 CFR 20 will be consulted.

2. Determination of Personnel Radiation Exposure, RP-1210-1

All personnel who have been issued a film badge at Zion Station will have either an active or inactive Form NRC-4 file.

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.102.

REFERENCES

FSAR Sections

12.3.3.3 Dosimeters

12.3.3.4 Monitoring of Visitors

Zion Procedures

RP-1210-1	Determination of Personnel Radiation Exposure
RP-1210-2	Film Badge Program

10 CFR 20.103 - EXPOSURE OF INDIVIDUALS TO CONCENTRATIONS OF RADIOACTIVE MATERIALS IN AIR IN RESTRICTED AREAS

STATEMENT OF SECTION 20.103 - PARAGRAPH (a)(1)

No licensee shall possess, use, or transfer licensed material in such a manner as to permit any individual in a restricted area to inhale a quantity of radioactive material in any period of one calendar quarter greater than the quantity which would result from inhalation for 40 hours per week for 13 weeks at uniform concentrations of radioactive material in air specified in Appendix B, Table I, Column 1. If the radioactive material is of such form that intake by absorption through the skin is likely, individual exposures to radioactive material shall be controlled so that the uptake of radioactive material by any organ from either inhalation or absorption or both routes of intake in any calendar quarter does not exceed that which would result from inhaling such radioactive material for 40 hours per week for 13 weeks at uniform concentrations specified in Appendix B, Table I, Column 1.

EVALUATION OF COMPLIANCE

1. Assessment of Exposure (MPC-Hour) To Radioactive Materials in Air, RP-1310-5

Individuals will not be permitted to receive ≥40 MPC-hours in any 7-day period.

2. Air Sampling and Posting of Suspected and Known Radioactive Airborne Areas, RP-1310-11

Normally air samples will be collected and analyzed prior to performing work in a controlled area. If air sample results cannot be obtained, the evaluation of continuous air monitors and past air sample data may be used to conservatively prescribe required respirators.

Air samples of previously posted airborne radioactivity areas should be obtained daily. If no entry into such an area is anticipated or the engineering controls of the area are such as to prevent exposure of individuals entering the area, no air sample need be obtained.

Air samples should be collected in close proximity of the worker to assure that a representative sample is obtained and that proper respirators are prescribed.

When active work is underway in a contaminated area, or high speed sawing, flame cutting, grinding, welding, or heating of contaminated material, a continuous air sample will be obtained as practicable.

Where the potential for airborne radioactivity exists, weekly air samples in the high traffic positions of the controlled area should be obtained.

STATEMENT OF SECTION 20.103 - PARAGRAPH (a)(2)

No licensee shall possess, use or transfer mixtures of U-234, U-235, and U-238 in soluble form in such a manner as to permit any individual in a restricted area to inhale a quantity of such material in excess of the intake limits specified in Appendix B, Table I, Column 1 of this part. If such soluble uranium is of a form such that absorption through the skin is likely, individual exposures to such material shall be controlled so that the uptake of such material by any organ from either inhalation or absorption or both routes of intake does not exceed that which would result from inhaling such material at the limits specified in Appendix B, Table I, Column 1 and footnote 4 thereto.

EVALUATION OF COMPLIANCE

Mixtures of U-234, U-235, and U-238 in soluble form are not present at Zion Station.

STATEMENT OF SECTION 20.103 - PARAGRAPH (a)(3)

For purposes of determining compliance with the requirements of this section the licensee shall use suitable measurements of concentrations of radioactive materials in air for detecting and evaluating airborne radioactivity in restricted areas and in addition, as appropriate, shall use measurements of radioactivity in the body, measurements of radioactivity excreted from the body, or any combination of such measurements as may be necessary for timely detection and assessment of individual intakes of radioactivity by exposed individuals. It is assumed that an individual inhales radioactive material at the airborne concentration in which he is present unless he uses respiratory protective equipment pursuant to paragraph (c) of this section. When assessment of a particular individual's intake of radioactive material is necessary, intakes less than those which would result from inhalation for 2 hours in any one day or for 10 hours in any one week at uniform concentrations specified in Appendix B, Table I, Column 1 need not be included in such assessment, provided that for any assessment in excess of these amounts the entire amount is included.

EVALUATION OF COMPLIANCE

1. Evaluation of Compliance to Section 20.103 - Paragraph (a)(1)

Evaluation of airborne activity in restricted areas is discussed in the evaluation of compliance to Section 20.103 - Paragraph (a)(1).

2. Personnel Bioassay Scheduling, RP-1340-2

Bioassays (in-vivo measurements; i.e., whole body counting, and/or measurements of radioactive material in excreta) will be conducted as necessary to aid in determining the extent of an individual's internal exposure to concentrations of radioactive material.

3. Issuance and Selection of Respiratory Equipment, RP-1310-2

Radiation protection personnel at Zion will evaluate the respiratory protection requirements for an area based on air sampling data and/or contamination surveys. Respiratory protection equipment will then be selected to provide a protection factor greater than the multiple by which peak concentrations of radioactive materials are expected to exceed the values specified in Table 1, Column 1 of Appendix B to 10 CFR 20.

4. Assessment of Exposure (MPC-hour) to Radioactive Materials, RP-1310-5

Radioactive material intakes >2 MPC-hours per day will be recorded on an individual's MPC-hour log. Such logs will be initiated and maintained on all individuals once they receive 2 MPC-hours in any 1 day.

In cases where the recorded MPC-hour exposure exceeds 5 MPC-hours in any 7-day period, the respitatory equipment log will be reviewed to assess the entire exposure for this period. If this assessment shows a 7-day exposure of 10 MPC-hours or more, the entire amount will be recorded on the MPC-hour log.

The respiratory equipment log contains the essential raw data (respirator user's name, type of respirator worn, airborne conditions, and time of respirator possession) in order to calculate the MPC-hours incurred by individuals wearing respirators.

STATEMENT OF SECTION 20.103 - PARAGRAPH (b)(1)

The licensee shall, as a precautionary procedure, use process or other engineering controls, to the extent practicable, to limit concentrations of radioactive materials in air to levels below those which delimit an airborne radioactivity area as defined in Section 20.203 - Paragraph (d)(1)(ii).

EVALUATION OF COMPLIANCE

The containment ventilation system is designed to provide a means to reduce the concentration of particulate and gaseous contamination to assure safe continuous access (40 hours/week) during normal reactor shutdown (FSAR Section 9.10.6, page 9.10-20).

Portable ventillation systems, hoods, and tests are used as practicable to contain and reduce airborne particulate and gaseous contamination during the performance of various jobs.

STATEMENT OF SECTION 20.103 - PARAGRAPH (b)(2)

When it is impracticable to apply process or other engineering controls to limit concentrations of radioactive material in air below those defined in Section 20.203 - Paragraph (d)(1)(ii), other precautionary procedures, such as increased surveillance, limitation of working times, or provisions of respiratory protective equipment, shall be used to maintain intake of radioactive material by any individual within any period of seven consecutive days as far below that intake of radioactive material which would result from inhalation of such material for 40 hours at the uniform concentrations specified in Appendix B, Table I, Column 1 as is reasonably achievable. Whenever the intake of radioactive material by any individual exceeds this 40-hour control measure, the licensee shall make such evaluations and take such actions as are necessary to assure against recurrence. The licensee shall maintain records of such occurrences, evaluations, and actions taken in a clear and readily identifiable form suitable for summary review and evaluation.

EVALUATION OF COMPLIANCE

1. Evaluation of Compliance to Section 20.103 - Paragraph (a)(1)

Evaluation of airborne activity in restricted areas is discussed in the evaluation of compliance to Section 20.103 - Paragraph (a)(1).

2. Issuance and Selection of Respiratory Equipment, RP-1310-2

Radiation protection personnel at Zion will select appropriate respiratory equipment so that contaminant concentration inhaled by the wearer does not exceed the appropriate regulatory limits specified in Appendix B, Table I, Column 1.

Should an individual receive greater than 40 MPC hours in 7 consecutive days, an evaluation will be made to identify the cause and actions will be taken to prevent recurrence. Records will be maintained for each occurence.

STATEMENT OF SECTION 20.103 - PARAGRAPH (c)

When respiratory protective equipment is used to limit the inhalation of airborne radioactive material pursuant to paragraph (b)(2) of this section, the licensee may make allowar e for such use in estimating exposures of individuals to such ma.erials provided that such equipment is used as stipulated in Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection."

EVALUATION OF COMPLIANCE

Radiation protection personnel at Zion will select appropriate respiratory equipment so that contaminant concentration inhaled by the wearer does not exceed the appropriate regulatory limits (Issuance and Selection of Respiratory Equipment, RP-1310-2).

The protection factors used at Zion Station comply with the protection factors permitted under Regulatory Guide 8.15 (Issuance and Selection of Respiratory Equipment, RP-1310-2).

The requirements of Regulatory Guide 8.15, "Acceptable Programs of Respiratory Protection," are encompassed throughout about 21 radiation protection procedures at Zion Station.

STATEMENT OF SECTION 20.103 - PARAGRAPHS (e) AND (f)

(e) The licensee shall notify, in writing, the Director of the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office listed in Appendix D at least 30 days before the date that respiratory protective equipment is first used under the provisions of this section.

(f) A licensee who was authorized to make allowance for use of respiratory protective equipment prior to December 29, 1976 shall bring his respiratory protective program into conformance with the requirements of paragraph (c) of this section within one year of that date, and is exempt from the requirement of paragraph (e) of this section.

EVALUATION OF COMPLIANCE

Zion Station is in compliance with 10 CFR 20.103 - Paragraph (c) and therefore is exempt from the requirements of Paragraph (e).

CONCLUSIONS

Zion Station complies with the intent of 10 CFR 20.103.

REFERENCES

FSAR Sections

9.10.6 Normal Containment Ventilation

FSAR Questions

9.8 Charcoal Filter Design

Zion Procedures

RP-1310-. Issuance and Selection of Respiratory Equipment

RP-1310-5	Assessment of Exposure [MPC-hour] to Radioactive Materials
RP-1310-11	Air Sampling and Posting of Suspected and Known Radioactive Airborne Areas
RP-1340-2	Personnel Bioassay Scheduling

10 CFR 20.104 - EXPOSURE OF MINORS

STATEMENT OF SECTION 20.104

(a) No licensee shall possess, use or transfer licensed material in such a manner as to cause any individual within a restricted area who is under 18 years of age, to receive in any period of one calendar quarter from radioactive material and other sources of radiation in the licensee's possession a dose in excess of 10 percent of the limits specified in the table in paragraph (a) of Section 20.101.

(b) No licensee shall possess, use or transfer licensed material in such a manner as to cause any individual within a restricted area, who is under 18 years of age to be exposed to airborne radioactive material possessed by the licensee in an average concentration in excess of the limits specified in Appendix B, Table II of this part. For purposes of this paragraph, concentrations may be averaged over periods not greater than a week.

(c) The provisions of Section 20.103 - Paragraphs (b)(2) and (c) shall apply to exposures subject to paragraph (b) of this section except that the references in Section 20.103 - Paragraphs (b)(2) and (c) to Appendix B, Table I, Column 1 shall be deemed to be references to Appendix B, Table II, Column 1.

EVALUATION OF COMPLIANCE

The maximum allowable quarterly radiation dose limits for a minor are 125 mrem-whole body, 750 mrem-skin, 1875 mrem-extremities (Exposure Control, RP-1210-5). Minors are only brought into control areas as part of a tour group. The requirements of 10 CFR 20.104 - Paragraphs (b) and (c) are met through adherence to Zion Station procedures (RP-1210-2, Film Badge Program).

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.104.

REFERENCES

Zion Procedures

RP-1210-5	Exposure Control
RP-1190-1	Zion Radiation Protection General Procedures, Zion Nuclear Power Station Radiation Control Standards
RP-1210-2	Film Badge Program

10 CFR 20.105 - PERMISSIBLE LEVELS OF RADIATION IN UNRESTRICTED AREAS

STATEMENT OF SECTION 20.105

(a) There may be included in any application for a license or for amendment of a license proposed limits upon levels of radiation in unrestricted areas resulting from the applicant's possession or use of radioactive material and other sources of radiation. Such applications should include information as to anticipated average radiation levels and anticipated occupancy times for each unrestricted area involved. The Commission will approve the proposed limits if the applicant demonstrates that the proposed limits are not likely to cause any individual to receive a dose to the whole body in any period of one calendar year in excess of 0.5 rem.

(b) Except as authorized by the Commission pursuant to paragraph (a) of this section, no licensee shall possess, use or transfer licensed material in such a manner as to create in any unrestricted area from radioactive material and other sources of radiation in his possessions:

(1) Radiation levels which, if any individual were continuously present in the area, could result in his receiving a dose in excess of two millirems in any one hour, or

(2) Radiation levels which, if an individual were continuously present in the area, could result in his receiving a dose in excess of 100 millirems in any seven consecutive days.

EVALUATION OF COMPLIANCE

The Zion Radiation Control Standards prohibit the spread of contamination outside of control areas. The standards also prohibit the creating of a Radiation Area in unprotected areas (RP-1190-1, Zion Radiation Protection General Procedures).

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.105.

REFERENCES

Zion Procedures

RP-1190-1 Zion Radiation Protection General Procedures, Zion Nuclear Power Station Radiation Control Standards

10 CFR 20.106 - RADIOACTIVITY IN EFFLUENTS TO UNRESTRICTED AREAS

STATEMENT OF SECTION 20.106 - PARAGRAPHS (a) AND (d)

(a) A licensee shall not possess, use, or transfer licensed material so as to release to an unrestricted area radioactive material in concentrations which exceed the limits specified in Appendix "B", Table II of this part, except as authorized pursuant to Section 20.302 or paragraph (b) of this section. For purposes of this section concentrations may be averaged over a period not greater than one year.

(d) For the purposes of this section the concentration limits in Appendix "B", Table II of this part shall apply at the boundary of the restricted area. The concentration of radioactive material discharged through a stack, pipe or similar conduit may be determined with respect to the point where the material leaves the conduit. If the conduit discharges within the restricted area, the concentration at the boundary may be determined by applying appropriate factors for dilution, dispersion, or decay between the point of discharge and boundary.

EVALUATION OF COMPLIANCE

1. Appendix I Report

Radioactivity in effluents released to unrestricted areas at Zion Station Units 1&2 is documented in the report "Information Relevant to Keeping Levels of Radioactivity in Effluents To Unrestricted Areas As Low As Is Reasonably Achievable" June 4, 1976 and Amendment 1, November 12, 1976. This report demonstrated compliance to 10 CFR 50 Appendix I.

2. Zion Station Radiological Safety Technical Specifications, Sections 3.11 and 3.12

Sections 3.11 and 3.12 of Zion Station Radiological Safety Technical Specifications provide the limits and condition for discharging radioactive liquids and gases from the Zion Station so that 10 CFR 20 requirements are not exceeded.

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.106.

REFERENCES

Reports

"Information Relevant to Keeping Levels of Radioactivity in Effluents To Unrestricted Areas As Low As Reasonably Achievable," June 4, 1976 and Amendment 1, November 12, 1976 (Appendix I Report).

Zion Station Radiological Safety Technical Specifications

3.11 Radioactive Liquids

3.12 Radioactive Gases

10 CFR 20.202 - PERSONNEL MONITORING

STATEMENT OF SECTION 20.202 - PARAGRAPH (a)

Each licensee shall supply appropriate personnel monitoring equipment to, and shall require the use of such equipment by:

(1) Each individual who enters a restricted area under such circumstances that he receives, or is likely to receive, a dose in any calendar quarter in excess of 25 percent of the applicable value specified in paragraph (a) of Section 20.101.

(2) Each individual under 18 years of age who enters a restricted area under such circumstances that he receives, or is likely to receive, a dose in any calendar quarter in excess of 5 percent of the applicable value specified in paragraph (a) of Section 20.101.

(3) Each individual who enters a high radiation area.

EVALUATION OF COMPLIANCE

A film badge is issued to all personnel who are permanently assigned to Zion Station.

The requirements of 10 CFR 20.202 are met by Zion radiation procedures (RP-1190-1, Zion Radiation Protection General Procedures, and RP-1210-2, Film Badge Program).

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.202.

REFERENCES

Zion Procedures

RP-1190-1 Zion Radiation Protection Control Procedures, Zion Nuclear Power Station Radiation Control Standards

RP-1210-2 Film Badge Program

10 CFR 20.203 - CAUTION SIGNS, LABELS, SIGNALS, AND CONTROLS

STATEMENT OF SECTION 20.203 - PARAGRAPH (a)(1)

Except as otherwise authorized by the Commission, symbols prescribed by this section shall use the conventional radiation caution colors (magenta or purple on yellow background). The symbol prescribed by this section is the conventional three-bladed design:

EVAULATION OF COMPLIANCE

All signs designating a radiation zone shall have a yellow background with magenta or purple lettering. At least one conventional magenta-colored, three-bladed radiation symbol must appear on each sign (Zion Radiation Protection General Procedures, RP-1190-1).

STATEMENT OF SECTION 20.203 - PARAGRAPH (a)(2)

In addition to the contents of signs and labels prescribed in this section, licensees may provide on or near such signs and labels any additional information which may be appropriate in aiding individuals to minimize exposure to radiation or to radioactive material.

EVALUATION OF COMPLIANCE

Additional information may be provided on or near such signs in order to aid an individual to minimize his exposure to radiation or radioactive materials (Zion Radiation Protection General Procedures, RP-1190-1).

STATEMENT OF SECTION 20.103 - PARAGRAPH (b)

Radiation Areas. Each radiation area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION (OR DANGER)
RADIATION AREA

EVALUATION OF COMPLIANCE

Each radiation area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words CAUTION RADIATION AREA (Zion FSAR Question 11.28, page Q11.28-1, and Zion Nuclear Power Station Radiation Control Standards, Section 2.b.

STATEMENT OF SECTION 20.203 - PARAGRAPH (c)(1)

High Radiation Areas. Each high radiation area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION (OR DANGER)
HIGH RADIATION AREA

EVALUATION OF COMPLIANCE

Each high radiation area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words DANGER-HIGH RADIATION AREA (Zion FSAR Question 11.28, page Q11.28-1, and Zion Nuclear Power Station Radiation Control Standards, Section 2.a.

STATEMENT OF SECTION 20.203 - PARAGRAPH (c)(2)

Each entrance or access point to a high radiation area shall be:

(i) Equipped with a control device which shall cause the level of radiation to be reduced below that at which an individual might receive a dose of 100 millirems in 1 hour upon entry into the area; or

(ii) Equipped with a control device which shall energize a conspicuous visible or audible alarm signal in such a manner that the individual entering the high radiation area and the licensee or a supervisor of the activity are made aware of the entry; or

(iii) Maintained locked except during periods when access to the area is required, with positive control over each individual entry.

EVALUATION OF COMPLIANCE

High radiation areas will have locked barriers which require a key and administrative approval for opening (FSAR, Question 11.28, page Q11.28-1). Administrative procedures in force at Zion Station also address Paragraph (c)(2).

STATEMENT OF SECTION 20.203 - PARAGRAPH (c)(3)

The controls required by subparagraph (2) of this paragraph shall be established in such a way that no individual will be prevented from leaving a high radiation area.

EVALUATION OF COMPLIANCE

Locks and keys are administratively controlled. The locks on high radiation area doors provide one-way control; i.e., they permit free egress from inside the room or enclosure.

STATEMENT OF SECTION 20.203 - PARAGRAPH (c)(4)

In the case of a high radiation area established for a period of 20 days or less, direct surveillance to prevent unauthorized entry may be substituted for the controls required by subparagraph (2) of this paragraph.

EVALUATION OF COMPLIANCE

Commonwealth Edison Company (CECo) presently complies with subparagraph (2) and will comply with subparagraph (4) for temporary high radiation areas as the need arises. CECo does acknowledge this requirement and will comply with it if the requirements of (2) cannot be met.

STATEMENT OF SECTION 20.203 - PARAGRAPH (c)(5)

Any licensee, or applicant for a license, may apply to the Commission for approval of methods not included in subparagraphs (2) and (4) of this paragraph for controlling access to high radiation areas. The Commission will approve the proposed alternatives if the licensee or applicant demonstrates that the alternative methods of control will prevent unauthorized entry into a high radiation area, and that the requirement of subparagraph (3) of this paragraph is met.

EVALUATION OF COMPLIANCE

Commonwealth Edison Company (CECo) presently complies with subparagraph (2) and will comply with subparagraph (4) for temporary high radiation areas as the need arises. CECo does acknowledge this requirement and will comply with it if the requirements of subparagraphs (2) and (4) cannot be met.

STATEMENT OF SECTION 20.203 - PARAGRAPH (c)(6)

Each area in which there may exist radiation levels in excess of 500 rems in one hour at one meter from a sealed radioactive source that is used to irradiate materials shall:

(i) Have each entrance or access point equipped with entry control devices which shall function automatically to prevent any individual from inadvertently entering the area when such radiation levels exist; permit deliberate entry into the area only after a control device is actuated that shall cause the radiation level within the area, from the sealed source, to be reduced below that at which it would be possible for an individual to recieve a dose in excess of 100 mrem in one hour; and prevent operation of the source if the source would produce radiation levels in the area that could result in a dose to an individual in excess of 100 mrem in one hour. The entry control devices required by this paragraph (c)(6) shall be established in such a way that no individual will be prevented from leaving the area.

(ii) Be equipped with additional control devices such that upon failure of the entry control devices to function as required by paragraph (c)(6)(i) of this section the radiation level within the area, from the sealed source, shall be reduced below that at which is would be possible for an individual to receive a dose in excess of 100 mrem in one hour; and visible and audible alarm signals shall be generated to make an individual attempting to enter the area aware of the hazard and the licensee or at least one other individual, who is familar with the activity and prepared to render or summon assistance, aware of such failure of the entry control devices.

(iii) Be equipped with control devices such that upon failure or removal of physical radiation barriers other than the source's shielded storage container the radiation level from the source shall be reduced below that at which it would be possible for an individual to receive a dose in excess of 100 mrem in one hour; and visible and audible alarm signals shall be generated to make potentially affected individuals aware of the hazards and the licensee or at least one other individual, who is familiar with the activity and prepared to render or summon assistance, aware of the failure or removal of the physical barrier. When the shield for the stored source is a liquid, means shall be provided to monitor the integrity of the sheild and to signal, automatically, loss of adequate shielding. Physical radiation barriers that comprise permanent structural components, such as walls, that have no credible probability of failure or removal in ordinary circumstances need not meet the requirements of this paragraph (c)(6)(iii).

(iv) Be equipped with devices that will automatically generate visible and audible alarm signals to alert personnel in the area before the source can be put into operation and in sufficient time for any individual in the area to operate a clearly identified control device which shall be installed in the area and which can prevent the source from being put into operation.

(v) Be controlled by use of such administrative procedure and such devices as are necessary to assure that the area is cleared of personnel prior to each use of the source preceeding which use it might have been possible for an individual to have entered the area.

(vi) Be checked by a physical radiation measurement to assure that prior to the first individual's entry into the area after any use of the source, the radiation level from the source in the area is below that at which it would be possible for an individual to receive a dose in excess of 100 mrem in one hour.

(vii) Have entry control devices required in paragraph(c)(6)(i) of this section which have been tested for proper functioning prior to initial operation with such source of radiation on any day that operations are not uninterruptedly continued from the previous day or before resuming operations after any unintended interruption, and for which records are kept of the dates, times, and results of such tests of function. No operations other than those necessary to place the source in safe condition or to effect repairs on controls shall be conducted with such source unless control devices are functioning properly. The licensee shall submit an acceptable schedule for more complete periodic tests of the entry control and warning systems to be established and adhered to as a condition of the license.

(viii) Have those entry and exit portals that are used in transporting materials to and from the irradiation area, and that are not intended for use by individuals, controlled by such devices and adminstrative procedures as are necessary to physically protect and warn against inadvertent entry by any individual through such portals. Exit portals for processed materials shall be equipped to detect and signal the presence of loose radiation sources that are carried toward such an exit and to automatically prevent such loose sources from being carried out of the area.

EVALUATION OF COMPLIANCE

Zion Station does not possess or intend to use such a source. Therefore, the requirements of this subparagraph are not applicable.

STATEMENT OF SECTION 20.203 - PARAGRAPH (c)(7)

Licensees with, or applicants for licenses for radiation sources that are within the purview of paragraph (c)(6) of this section, and that must be used in variety of positions or in peculiar locations, such as open fields or forests, that make it impracticable to comply with certain requirements of paragraph (c)(6) of this section, such as those for the automatic control of radiation levels, may apply to the Director, Office of Nuclear Material Safety and Safeguards, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, for approval prior to use of safety measures that are alternative to those specified in paragraph (c)(6) of this section, and that will provide at least an equivalent degree of personnel protection in the use of such sources. At least one of the alternative measures must include an entry-preventing interlock control based on a physical measurement of radiation that assures the absence of high radiation levels before an individual can gain access to an area where such sources are used.

EVALUATION OF COMPLIANCE

This requirement is not applicable to Zion Station.

STATEMENT OF SECTION 20.203 - PARAGRAPH (d)

Airborne Radioactivity Areas. (1) As used in the regulations in this Part, "airborne radioactivity area" means (i) any room, enclosure, or operating area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations in excess of the amounts specified in Appendix "B", Table I, Column 1 of this Part; or (ii) any room, enclosure, or operating area in which airborne radioactive material composed wholly or partly of licensed material exists in concentrations which, averaged over the number of hours in any week during which individuals are in the area, exceed 25% of the amounts specified in Appendix "B", Table I Column 1 of this part.

(2) Each airborne radioactivity area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION*
AIRBORNE RADIOACTIVITY AREA

EVALUATION OF COMPLIANCE

An area will be posted as an airborne radioactivity area during the following conditions (Air Sampling and Posting of Suspected and Known Radioactive Airborne Areas, RP-1310-11, and Zion Nuclear Power Station Radiation Control Standards, Section 2.b.

a. The isotopic particulate results are greater than 0.25 MPC as determined by gamma spectroscopy, or

b. The 6-hour gross β,γ results are greater than 7.5×10^{-10} μCi/cc or

c. The 24-hour alpha activity is greater than 2×10^{-12} μCi/cc

STATEMENT OF SECTION 20.203 - PARAGRAPH (e)

Additional Requirements: (1) Each area or room in which licensed material is used or stored and which contains any radioactive material (other than natural uranium or thorium) in an amount exceeding 10 times the quantity of such material specified in Appendix "C" of this Part shall be conspicously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION*
RADIOACTIVE MATERIAL(S)

(2) Each area or room in which natural uranium or thorium is used or stored in an amount exceeding one-hundred times the quantity specified in Appendix "C" of this Part shall be conspicously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION*
RADIOACTIVE MATERIAL(S)

EVALUATION OF COMPLIANCE

1. Movement of Radioactive Material, RP-1440-2

All material with greater than 100 cpm/ft^2 fixed or smearable activity will be marked as radioactive material. Radioactive material shall be placed in, or established as a radiation zone.

2. Zion Radiation Protection General Procedures, RP-1190-1

The wording on the sign used for posting a radiation zone composed of radioactive materials is CAUTION - RADIOACTIVE MATERIALS - AUTHORIZED ENTRY ONLY. Applicable combinations of wording may be used on the same signs; i.e., CAUTION - RADIATION AREA - RADIOACTIVE MATERIALS - AUTHORIZED ENTRY ONLY.

STATEMENT OF SECTION 20.203 - PARAGRAPH (f)(1)

Containers. (1) Except as provided in subparagraph (3) of this paragraph, each container of licensed material shall bear a durable, clearly visible label identifying the radioactive contents.

EVALUATION OF COMPLIANCE

1. Zion Radiation Protection General Procedures, RP-1190-1

Containers intended for offsite disposal must be numbered, weighed, surveyed, and labelled when placed in the temporary storage area.

2. Inventory and Leak Test of Radiation Sources, RP-1480-1

Each container of nonexempt radioactive sources shall have a durable, clearly visible label identifying the radioactive contents.

STATEMENT OF SECTION 20.203 - PARAGRAPH (f)(2)

A label required pursuant to subparagraph (1) of this paragraph shall bear the radiation caution symbol and the words "CAUTION, RADIOACTIVE MATERIAL" or "DANGER, RADIOACTIVE MATERIAL". It shall also provide sufficient information to permit individuals handling or using the containers, or working in the vicinity thereof, to take precautions to avoid or minimize exposures.

EVALUATION OF COMPLIANCE

Radioactive material labels shall have a yellow background with magenta or purple lettering and at least one conventional magenta three-bladed radiation symbol on each label (Zion Radiation Protection General Procedures RP-1190-1).

Space is provided on the label for insertion of an explantory message (Zion Radiation Protection General Procedures, RP-1190-1).

STATEMENT OF SECTION 20.203 - PARAGRAPH (f)(3)

Nowithstanding the provisions of subparagraph (1) of this paragraph, labeling is not required:

(i) For containers that do not contain licensed materials in quantities greater than the applicable quantities listed in Appendix C of this part.

(ii) For containers containing only natural uranium or thorium in quantities no greater than 10 times the applicable quantities listed in Appendix C of this part.

(iii) For containers that do not contain licensed materials in concentrations greater than the applicable concentrations listed in Column 2, Table I, Appendix B of this part.

(iv) For containers when they are attended by an individual who takes the precautions necessary to prevent the exposure of any individual to radiation or radioactive materials in excess of the limits established by the regualtions in this part.

(v) For containers when they are in transport and packaged and labeled in accordance with regulations of the Department of Transportation.

(vi) For containers which are accessible only to individuals authorized to handle or use them, or to work in the vicinity thereof, provided that the contents are identified to such individuals by a readily available written record.

(vii) For manufacturing or process equipment, such as nuclear reactors, reactor components, piping, and tanks.

EVALUATION OF COMPLIANCE

Zion Station recognises these exceptions to the labelling requirements for containers.

STATEMENT OF SECTION 20.203 - PARAGRAPH (f)(4)

Each licensee shall, prior to disposal of an empty container to unrestricted areas, remove or deface the radioactive material label or otherwise clearly indicate that the container no longer contains radioactive materials.

EVALUATION OF COMPLIANCE

Signs should warn of an existing radiological hazard and should be promptly removed when no longer required (Zion Radiation Protection General Procedures, RP-1190-1).

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.203.

REFERENCES

FSAR Questions

11.28 Radiation Areas

Zion Procedures

RP-1190-1	Zion Radiation Protection General Procedures
RP-1310-11	Air Sampling and Posting of Suspected and Known Radioactive Airborne Areas
RP-1440-2	Movement of Radioactive Material
RP-1480-1	Inventory and Leak Test of Radioactive Sources

10 CFR 20.205 - PROCEDURES FOR PICKING UP, RECEIVING, AND OPENING PACKAGES

STATEMENT OF SECTION 20.205 - PARAGRAPH (a)(1)

Each licensee who expects to receive a package containing quantities of radioactive material in excess of the Type A quantities shall:

(i) If the package is to be delivered to the licensee's facility by the carrier, make arrangements to receive the package when it is offered for delivery by the carrier; or

(ii) If the package is to be picked up by the licensee at the carrier's terminal, make arrangements to receive notification from the carrier of the arrival of the package, at the time of arrival.

EVALUATION OF COMPLIANCE

Personnel ordering radioactive material or equipment containing radioactive material should inform the Radiation Protection Supervisor and the intended recipient, if other then himself, of the nature of the material and the expected date of arrival (Zion Radiation Protection General Procedures, RP-1190-1 and RP-1530-2, Receipt of Radioactive Material).

STATEMENT OF SECTION 20.205 - PARAGRAPH (a)(2)

Each licensee who picks up a package of radioactive material from a carrier's terminal shall pick up the package expeditiously upon receipt of notification from the carrier of its arrival.

EVALUATION OF COMPLIANCE

A radioactive material shipment to be picked up from a carrier's terminal shall be made as expenditiously as possible upon notification by the carrier (Receipt of Radioactive Material, RP-1530-2).

STATEMENT OF SECTION 20.205 - PARAGRAPH (b)(1) (SUMMARY)

Each licensee, upon receipt of a package of radioactive material, shall monitor the external surfaces of the package for radioactive contamination caused by leakage of the radioactive contents, except as noted in this paragraph.

The monitoring shall be performed as soon as practicable after receipt, but no later than three hours after the package is received at the licensee's facility if received during the licensee's normal working hours, or eighteen hours if received after normal working hours.

EVALUATION OF COMPLIANCE

Upon arrival of the shipment, radiation protection personnel should be notified immediately so that surveys can be made of the vehicle transporting the radioactive material and the package containing the material (Receipt of Radioactive Material, RP-1530-2).

STATEMENT OF SECTION 20.205 - PARAGRAPH (b)(2)

If removable radioactive contamination in excess of 0.01 microcuries (22,000 disintegration per minute) per 100 square centimeters of package surface is found on the external surfaces of the package, the licensee shall immediately notify the final delivering carrier and, by telephone and telegraph, mailgram, or facsimile, the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office shown in Appendix D.

EVALUATION OF COMPLIANCE

The shipment should be surveyed before unloading to assess contamination levels. Notify radiation protection supervision immediately if the following conditions occur (Receipt of Radioactive Material, RP-1530-2):

1. The average smearable beta-gamma contamination exceeds 2200 dpm per 100 cm^2;

2. The average smearable alpha contamination exceeds 220 dpm per 100 cm^2.

All 10 CFR 20 reporting requirements are complied with per existing procedures.

STATEMENT OF SECTION 20.203 - PARAGRAPH (c)(1)

Each licensee, upon receipt of a package containing quantities of radioactive material in excess of the Type A quantities specified in paragraph (b) of this section, other than those transported by exclusive use vehicle, shall monitor the radiation levels external to the package. The package shall be monitored as soon as practicable after receipt, but no later than three hours after the package is received at the licensee's facility if received during the licensee's normal working hours, or 18 hours if received after normal working hours.

EVALUATION OF COMPLIANCE

Upon arrival of the shipment, radiation protection personnel should be notified immediately so that surveys can be made of the vehicle transporting the material and the package containing the material (Receipt of Radioactive Material, RP-1530-2).

STATEMENT OF SECTION 20.205 - PARAGRAPH (c)(2)

If radiation levels are found on the external surface of the package in excess of 200 millirem per hour, or at three feet from the external surface of the package in excess of 10 millirem per hour, the licensee shall immediately notify, by telephone and telegraph, mailgram, or facsimile, the Director of the appropriate NRC Regional Office listed in Appendix D, and the final delivering carrier.

EVALUATION OF COMPLIANCE

The shipment should be surveyed before unloading to assess exposure rates. Notify radiation protection supervision if the exposure rate exceeds 200 millirem per hour at contact with the package and/or 10 millirem per hour 3 feet from the package (Receipt of Radioactive Material, RP-1530-2).

All 10 CFR 20 reporting requirements are complied with per existing procedures.

STATEMENT OF SECTION 20.205 - PARAGRAPH (d)

Each licensee shall establish and maintain procedures for safely opening packages in which licensed material is received, and shall assure that such procedures are followed and that due consideration is given to special instructions for the type of package being opened.

EVALUATION OF COMPLIANCE

Before any package containing licensed material is opened (Zion Radiation Protection General Procedures, RP-1190-1):

1. Radiation protection personnel will read the radioactive shipment record to assess radiological hazards, and

2. Radiation protection personnel will determine if a special work permit (SWP) will be required.

If an SWP is required, it will be executed as per the special work permit section of the radiation control standards.

If an SWP is not required by radiation protection personnel after consideration of the radioactive shipment record, oral or written procedures will be issued if deemed necessary (Zion Radiation Protection General Procedures, RP-1190-1).

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.205.

REFERENCES

Zion Procedures

RP-1190-1 Zion Radiation Protection General Procedures

RP-1530-2 Receipt of Radioactive Materials

10 CFR 20.206 - INSTRUCTION OF PERSONNEL

STATEMENT OF SECTION 20.206

Instructions required for individuals working in or frequenting any portion of a restricted area are specified in Section 19.12 of this chapter (which is provided below).

STATEMENT OF SECTION 19.12

Instructions to workers. All individuals working or frequenting any portion of a restricted area shall be kept informed of the storage, transfer, or use of radioactive materials or of radiation in such portions of the restricted area; shall be instructed in the health protection problems associated with exposure to such radioactive materials or radiation, in precautions or procedures to minimize exposure, and in the purposes and functions of protection devices employed; shall be instructed in, and instructed to observe, to the extent within the worker's control, the applicable provisions of Commission regulations and licenses for the protection of personnel from exposure to radiation or radioactive materials occuring in such areas; shall be instructed of their responsibility to report promptly to the licensee any condition which may lead to or cause a violation of Commission regulations and licenses or unnecessary exposure to radiation or to radioactive material; shall be instructed in the appropriate response to warnings made in the event of any unusual occurrence or malfunction that may involve exposure to radiation or radioactive material; and shall be advised as to the radiation exposure reports which workers may request pursuant to Section 19.13. The extent of these instructions shall be commensurate with potential radiological health protection problems in the restricted area.

EVALUATION OF COMPLIANCE

1. FSAR, Section 12.3.3, page 12.3-5

Each individual is trained to minimize his exposure consistent with discharging his duties. Each individual is responsible for observing rules adopted for his safety and that of others.

Radiation protection personnel evaluate radiological conditions of operations and establish the procedures to be followed by all personnel. They ensure that all applicable regulations are complied with and that the required radiation protection records are adequately maintained.

2. Film Badge Program, RP-1210-2

All individuals receiving film badges must have received radiation training conmensurate with the potential radiological health protection problems associated with the work assignment.

3. FSAR, Section 12.3.3.5, page 12.3-6

All personnel entering a radioactive materials area are required to wear the protective clothing specified by radiation protection personnel. The clothing requirements are established based on evaluation of the radiological conditions of the area.

4. FSAR, Section 12.3.2, page 12.3-4

If it is determined by fixed and/or portable radiation monitoring devices that radiation from or within the station is such that permissible exposures in restricted and unrestricted areas will be exceeded if occupancy of these areas is continued, the evacuation alarm is sounded, the unit is shut down, and all personnel not essential to the emergency shutdown procedures immediately assemble at a safe location in accordance with Zion Station emergency procedures.

5. Evaluation of Compliance for Section 20.409

The evaluation of 10 CFR Section 20.409 discusses the notification of individuals pursuant to 10 CFR 19.13.

6. Zion Radiation Control Standards, Work in Control Areas Section

Guidance for work performed in controlled areas is found in the Zion Nuclear Power Station Radiation Control Standards, Work in Control Areas Section.

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.206.

REFERENCES

FSAR Sections

12.3.2.6 High Radiation Evacuation

12.3.3 Radiation Control Standards and Procedures

12.3.3.5 Personnel Protection Equipment

Zion Procedures

Zion Nuclear Power Station Radiation Control Standards

RP-1210-2 Film Badge Program

10 CFR 20.207 - STORAGE AND CONTROL OF LICENSED MATERIALS IN UNRESTRICTED AREAS

STATEMENT OF SECTION 20.207

(a) Licensed materials stored in an unrestricted area shall be secured from unauthorized removal from the place of storage.

(b) Licensed materials in an unrestricted area and not in storage shall be tended under the constant surveillance and immediate control of the licensee.

EVALUATION OF COMPLIANCE

Licensed materials are not stored outside of the restricted area at Zion.

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.207.

10 CFR 20.301 - WASTE DISPOSAL GENERAL REQUIREMENT

STATEMENT OF SECTION 20.301

No licensee shall dispose of licensed material except:

(a) By transfer of an authorized recipient as provided in the regulations in Part 30, 40, or 70 of this Chapter, whichever may be applicable; or

(b) As authorized pursuant to Section 20.302; or

(c) As provided in Section 20.303 or 20.304, applicable respectively to the disposal of licensed material by release into sanitary sewage systems or burial in soil, or in Section 20.106 (Radioactivity in effluents to unrestricted areas).

EVALUATION OF COMPLIANCE

Zion Procedure RP-1520-1, "Offsite Shipment of Radioactive Material," requires Zion Station to verify that the consignee is licensed to receive the type and amount of radioactive material being shipped.

Paragraph (b) is not applicable for Zion Station.

Sections 20.303 and 20.304 pertain to sewers and burial of radioactive waste, respectively, and, therefore, are not applicable for evaluation. The evaluation of compliance for Section 20.106 is presented on page 20.106-1.

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.301.

REFERENCES

Zion Procedures

RP-1520-1 Offsite Shipment of Radioactive Material

10 CFR 20.303 - DISPOSAL BY RELEASE INTO SANITARY SEWERAGE SYSTEMS

STATEMENT OF SECTION 20.303

No licensee shall discharge licensed material into a sanitary sewarage system unless:

(a) It is readily soluble or dispersible in water; and

(b) The quantity of any licensed or other radioactive material released into the system by the licensee in any one day does not exceed the larger of subparagraphs (1) or (2) of this paragraph:

> (1) The quantity which, if diluted by the average daily quantity of sewage released into the sewer by the licensee, will result in an average concentration equal to the limits specified in Appendix "B", Table I, Column 2 of this Part; or
>
> (2) Ten times the quanity of such material specified in Appendix "C" of this Part;

(c) The quantity of any licensed or other radioactive material released in any one month, if diluted by the average monthly quantity of water released by the licensee, will not result in an average concentration exceeding the limits specified in Appendix "B", Table I, Column 2 of this Part; and

(d) The gross quantity of licensed and other radioactive material released into the sewerage system by the licensee does not exceed one curie per year.

Excreta from individuals undergoing medical diagnosis or therapy with radioactive material shall be exempt from any limitations contained in this section.

EVALUATION OF COMPLIANCE

Radioactive water at Zion Station Units 1 & 2 is not released into the sanitary sewage system.

CONCLUSION

Not applicable.

REFERENCES

FSAR Sections

11.1.2 (Waste Disposal System) System Design Operation

10 CFR 20.304 - DISPOSAL BY BURIAL IN SOIL

STATEMENT OF SECTION 20.304

(a) The total quantity of licensed and other radioactive materials buried at any one location and time does not exceed at the time of burial, 1,000 times the amount specified in Appendix "C" of this Part; and

(b) Burial is at a minimum depth of four feet; and

(c) Successive burials are separated by distances of at least six feet and not more than 12 burials are made in any year.

EVALUATION OF COMPLIANCE

Radioactive waste at Zion Station Units 1 & 2 is not disposed of by burial at the Zion site.

CONCLUSION

Not applicable.

REFERENCES

FSAR Section

11.1.2 (Waste Disposal System) System Design and Operation

10 CFR 20.401 - RECORDS OF SURVEYS, RADIATION MONITORING, AND DISPOSAL

STATEMENT OF 20.401 - PARAGRAPH (a)

Each licensee shall maintain records showing the radiation exposures of all individuals for whom personnel monitoring is required under paragraph 20.202 of the regulations in this part. Such records shall be kept on Form NRC-5 in accordance with the instructions contained in that form or on clear and legible records containing all the information required by Form NRC-5. The doses entered on the forms or records shall be for periods of time not exceeding one calender quarter.

EVALUATION OF COMPLIANCE

The personnel occupational dose exposure records at Zion Station are maintained on microfiche making use of a computerized dosimetry program (TSN-CR-D9 Microfiche Output of Bi-Weekly Computerized Dosimetry Reports, NRC Form-5).

STATEMENT OF SECTION 20.401 - PARAGRAPH (b)

Each licensee shall maintain records in the same units used in this part showing the results of surveys required by 20.201(b); monitoring required by 20.205(b) and 20.205(c); and disposals made under 20.302, 20.303, and 20.304.

EVALUATION OF COMPLIANCE

See evaluations of the sections that are referenced in Section 20.401(b).

STATEMENT OF SECTION 20.401 - PARAGRAPH (c)(1)

Records of individual exposure to radiation and to radioactive material which must be maintained pursuant to the provisions of paragraph (a) of this section and records of bioassays, including results of whole body counting examinations, made pursuant to 20.108, shall be preserved until the Commission authorizes disposition.

EVALUATION OF COMPLIANCE

Whole body counts are done on a routine schedule and recorded on the workers permanent NRC-Form 5. NRC Form-5 is preserved until the Commission authorizes disposition (Zion Procedure RP-1340-2 and Technical Services Nuclear Procedure TSN-CR-D4).

STATEMENT OF SECTION 20.401 - PARAGRAPH (c)(2)

Records of the results of surveys and monitoring which must be maintained pursuant to paragraph (b) of this section shall be preserved for two years after completion of the survey except that the following records shall be maintained until the Commission authorizes their disposition: (i) records of the results of surveys to determine compliance with 20.103 (a); (ii) in the absence of personnel monitoring data, records of the results of surveys to determine external radiation dose; and (iii) records of the results of surveys used to evaluate the release of radioactive effluents to the environment.

EVALUATION OF COMPLIANCE

Routine survey and monitoring records are kept on file or microfilmed until the Commission authorizes their disposition (Zion Procedure RP-1280, Series Surveys). In the absence of personnel monitoring data, records of the results of surveys to determine external radiation dose become part of a worker's permanent file (Zion Procdure RP-1230-1).

STATEMENT OF SECTION 20.401 - PARAGRAPH (c)(3)

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Records of disposal of licensed material made pursuant to Sections 20.302, 20.303, or 20.304 shall be maintained until the Commission authorizes their dispostion.

EVALUATION OF COMPLIANCE

Sources are inventoried and handled according to Zion Procedure RP 1480-1. Records are filmed or maintained on hard copy.

STATEMENT OF SECTION 20.401 - PARAGRAPHS (c)(4) and (c)(5)

(4) Records which must be maintained pursuant to this part may be the original or a reproduced copy or microform if such reproduced copy or microform is duly authenticated by authorized personnel and the microform is capable of producing a clear and legible copy after storage for the period specified by Commission regulations.

(5) If there is a conflict between the Commission's regulations in this part, license conditions, or technical specification, or other written Commission approval or authorization pertaining to the retention period for the same type of record, the retention period specified in the regulations in this part for such records shall apply unless the Commission, pursuant to 20.501, has granted a specific exemption from the record retention requirements specified in the regulations in this part.

EVALUATION OF COMPLIANCE

Records are filmed and stored in compliance with these regulations.

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.401.

REFERENCES

Zion Procedures

RP-1190-1	Radiation Control Standards
RP-1230-1	Lost Film Badge Procedure
RP-1280	Series Surveys
RP-1360	Series Survey Schedule
RP-1480-1	Source Inventory and Surveys
RP-1340-2	Personnel Bioassay Schedule

Technical Services Nuclear Procedures

TSN-CR-D4	Instructions for Entering Whole Body Data to Dosimetry Program
TSN-CR-D9	Microfiche Output Bi-Weekly Computerized Dosimetry Reports (NRC Form-5)

10 CFR 20.402 - REPORTS OF THEFT OR LOSS OF LICENSED MATERIAL

STATEMENT OF SECTION 20.402

(a) Each licensee shall report by telephone to the Director of the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office listed in Appendix D, immediately after its occurrence becomes known to the licensee, any loss or theft of licensed material in such quantites and under such circumstances that it appears to the licensee that a substantial hazard may result to persons in unrestricted areas.

(b) Each licensee who is required to make a report pursuant to paragraph (a) of this section shall, within thirty (30) days after he learns of the loss or theft, make a report in writing to the appropriate NRC Regional Office listed in Appendix D with copies to the Director of Inspection and Enforcement, U.S. Nuclear Regulation Commission, Washington, D.C. 20555, setting forth the following information:

(1) A description of the licensed material involved, including kind, quantity, chemical, and physical form;

(2) A description of the circumstances under which the loss or theft occurred;

(3) A statement of disposition or probable disposition of the licensed material involved; $/:';08;#a=3 (4) Radiation exposures to individuals, circumstances under which the exposures occurred, and the extent of possible hazard to persons in unrestricted areas;

(5) Actions which have been taken, or will be taken, to recover the material; and

(6) Procedures or measures which have been or will be adopted to prevent a recurrence of the loss or theft of licensed material.

(c) Subsequent to filing the written report the licensee shall also report any substantive additional information on the loss or theft which becomes available to the licensee, within 30 days after he learns of such information.

(d) Any report filed with the Commission pursuant to this section shall be so prepared that names of individuals who may have received exposure to radiation are stated in a separate part of the report.

EVALUATION OF COMPLIANCE

The disposal and inventory of radiation sources is as stated in Zion Procedures RP-1486-1, RP-1610-1, and RP-1610-2.

CONCLUSION

Zion Station is in compliance with 10 CFR 20.402.

REFERENCES

Zion Procedures

RP-1486-1 Source Inventory and Surveys

RP-1610-1 Disposal of Radioactive Sources

RP-1610-2 Radioactive Sources

10 CFR 20.403 - NOTIFICATION OF INCIDENTS

STATEMENT OF SECTION 20.403 - PARAGRAPH (a)

Immediate notification. Each licensee shall immediately notify by telephone and telegraph, mailgram, or facsimile, the Director of the appropriate NRC Regional Office listed in Appendix D of any incident involving byproduct, source, or special nuclear material possessed by him and which may have caused or threatens to cause:

(1) Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body of any individual of 150 rems or more of radiation; or exposure of the feet, ankles, hands or forearms of any individual to 375 rems or more of radiation; or

(2) The release of radioactive material in concentration which, if averaged over a period of 24 hours, would exceed 5,000 times the limits specified for such materials in Appendix B, Table II; or

(3) A loss of one working week or more of the operation of any facilities affected; or

(4) Damage to property in excess of $200,000.

EVALUATION OF COMPLIANCE

In case of the incidents specified in Section 20.403 - Paragraphs (a)(1), (a)(2), (a)(3), and (a)(4), Zion Station provides immediate notification to the NRC (Technical Services Nuclear Procedure TSN-CR-G3).

STATEMENT OF SECTION 20.403 - PARAGRAPH(b)

Twenty-four hour notification. Each licensee shall within 24 hours notify by telephone and telegraph, mailgram, or facsimile, the Director of the appropriate NRC Regional Office listed in Appendix D of any incident involving licensed material possessed by him and which may have caused or threatens to cause:

(1) Exposure of the whole body of any individual to 5 rems or more radiation; exposure of the skin of the whole body of any individual to 30 rems or more of radiation; or exposure of the feet, ankles, hands, or forearms to 75 rems or more radiation; or

(2) The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 500 times the limits specified for such materials in Appendix B, Table II; or

(3) A loss of one day or more of the operation of any facilities affected; or

(4) Damage to property in excess of $2,000.

EVALUATION OF COMPLIANCE

In case of the incidents specified in Section 20.403 - Paragraphs (a)(1), (a)(2), (a)(3), and (a)(4), Zion Station provides immediate notification to the NRC (Technical Services Nuclear Procedure TSN-CR-G3).

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.403.

REFERENCES

Technical Services Nuclear Procedures

TSN-CR-G3	Compilation of Reporting Requirements for Persons Subject to NRC, State of Illinois, or State of Indiana Regulations for Protection Against Radiation.

10 CFR 20.405 - REPORTS OF OVEREXPOSURES AND EXCESSIVE LEVELS AND CONCENTRATIONS

STATEMENT OF SECTION 20.405 - PARAGRAPH (a)

(a) In addition to any notification required by 20.402, each licensee shall make a report in writing within 30 days to the appropriate NRC Regional Office listed in Appendix D with a copy to the Director of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, of:

(1) each exposure of an individual to radiation in excess of the applicable limits in Sections 20.101 or 20.104 (a) or the license; (2) each exposure of an individual to radioactive material in excess of the applicable limits in Sections 20.103 (a)(1), 20.103(a)(2), 20.104(b), or the license; (3) levels of radiation or concentrations of radioactive material in a restricted area in excess of any other applicable limit in the license; (4) any incident for which notification is required by Section 20.403; and (5) levels of radiation or concentration of radioactive material (whether or not involving excessive exposure of any individual) in an unrestricted area in excess of ten times any applicable limit set forth in this part or in the license. Each report required under this paragraph shall describe the extent of exposure of individuals to radiation or to radioactive material, including estimates of each individuals's exposure as required by paragraph (b) of this section; levels of radiation and concentrations of radioactive material involved; the cause of the exposure, levels or concentrations; and corrective steps taken or planned to assure against a recurrence.

EVALUATION OF COMPLIANCE

In the case of an overexposure or excessive levels and concentrations, notification is done in compliance with the specifications. Commonwealth Edison Radiation Control Standards require compliance with 10 CFR 20.

STATEMENT OF SECTION 20.405 - PARAGRAPH(b)

Any report filed with the Commission pursuant to this section shall include for each individual exposed the name, social security number, and date of birth; and an estimate of the individual's exposure. The report shall be prepared so that this information is stated in a separate part of the report.

EVLAUATION OF COMPLIANCE

All personnel exposure records in Form NRC-5 include all the data required by this section. Reports of personnel exposure are prepared using Form-5 data.

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.405.

REFERENCES

Commonwealth Edison Radiation Control Standards

10 CFR 20.407 - PERSONNEL MONITORING REPORTS

STATEMENT OF SECTION 20.407

Each person described in Section 20.408 of this part shall, within the first quarter of each calendar year, submit to the Director of Management and Program Analysis, U.S. NUclear Regulatory Commission, Washington, D.C. 2055, the reports specified in paragraphs (a) and (b) of this section covering the preceding calendar years. All other persons specifically licensed by the Commission shall, within the first quarter of calendar years 1979 and 1980, submit to the Director of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, the reports specified in paragraphs (a) and (b) of this section covering the preceding calendar years 1978 and 1979.

(a) A report of either (1) the total number of individuals for whom personnel monitoring was required under 20.202(a) or 34.33(a) of this chapter during the calendar year; or (2) the total number of individuals for whom personnel monitoring was provided during the calendar year: Provided, however, that such total includes at least the number of individuals required to be reported under paragraph (a)(1) of this section. The report shall indicate whether it is submitted in accordance with paragraphph (a)(1) or (a)(2) of this section. If personnel monitoring was not required to be provided to any individual by the licensee under 20.202(a) or 34.33(a) of this chapter during the calendar year, the licensee shall submit a negative report indicating that such personnel monitoring was not required.

(b) A statistical summary report of the personnel monitoring information recorded by the licensee for individuals for whom personnel monitoring was either required or provided as described in paragraph (a) of this section, indicating the number of individuals whose total whole body exposure recorded during the previous calendar year was in each of the following estimated exposure ranges:

Estimated Whole Body Exposure Range (REMS)*	Number of Individuals in Each Range
No measurable exposure	....
Measurable exposure less than 0.1 ...	....
0.1 to 0.25	....
0.25 to 0.5	....
0.5 to 0.75	....
0.75 to 1	....
1 to 2	....
2 to 3	....
3 to 4	....
4 to 5	....
6 to 7	....
7 to 8	....
8 to 9	....
9 to 10	....
10 to 11	....
11 to 12	....
12+	....

*Individual values exactly equal to the values separating exposure ranges shall be reported in the higher range.

The low exposure range data are required in order to obtain better information about the exposure actually recorded. This section does not require improved measurements.

EVALUATION OF COMPLIANCE

Commonwealth Edison provides the reports required for radiation exposure statistics at Zion Station. These statistics are in the format specified (Technical Services Nuclear Procedure TSN-CR-D19).

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.407.

REFERENCES

Technical Services Nuclear Procedures

TSN-CR-D19 Occupational Radiation Exposure Statistics

10 CFR 20.408 - REPORTS OF PERSONNEL MONITORING ON TERMINATION OF EMPLOYMENT

STATEMENT OF SECTION 20.408

(a) This section applies to each person licensed by the Commission to:

(1) Operate a nuclear reactor designed to produce electrical or heat energy pursuant to 50.21(b) or 50.22 of this chapter or a testing facility as defined in 50.2(r) of this chapter.

(2) Possess or use byproduct material for purposes of radiography pursuant to Parts 30 and 34 of this chapter;

(3) Possess or use at any one time, for purposes of fuel processing, fabrication, or reprocessing, special nuclear material in a quantity exceeding 5,000 grams of contained uranium-235, uranium-233, or plutonium or any combination thereof pursuant to Part 70 of this chapter; or

(4) Possess or use at any one time, for processing or manufacturing for distribution pursuant to part 30, 32, or 33 of this chapter, byproduct material in quantities exceeding any one of the following quantities.

Radionuclide*	Quantity in curies
Cesium-137	1
Cobalt-60	1
Gold-198	100
Iodine-131	1
Iridium-192	10
Krypton-85	1,000
Promethium-147	10
Technetium-99m	1,000

*The Commission may require, as a license condition, or by rule, regulation or order pursuant to Section 20.502, reports from licensees who are licensed to use radionulides not on this list, in quantities sufficient to cause comparable radiation levels.

(b) When an individual terminates employment with a licensee described in paragraph (a) of this section, or an individual assigned to work in such a licensee's facility but not employed by the licensee, completes the work assignment in the licensee's facility, the licensee shall furnish to the Director of Management and Progr am Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 2055, a report of the individual's exposures to radiation and radioactive material, incurred during the period of employment or work assignment in the licensee's facility, containing information recorded by the licensee pursuant to Sections 20.401(a) and 20.108. Such report shall be furnished within 30 days after the exposure of the individual has been determined by the licensee or 90 days after the date of termination of employment or work assignment, whichever is earlier.

EVALUATION OF COMPLIANCE

Commonwealth Edison maintains a computer program which generates termination letters for individuals who have terminated employment at any of the Commonwealth Edison nuclear stations. These letters contain the information specified in Section 20.408 and are gen 0 days after the employees assignment ends.

CONCLUSION

Zion Station complies with the intent of 10 CFR 20.408.

REFERENCES

Technical Services Nuclear Procedures

TSN-CR-D16 Instructions for Handling Computer Generated Personnel Dosimetry Termination Letters

10 CFR 20.409 - NOTIFICATIONS AND REPORTS TO INDIVIDUALS

STATEMENT OF SECTION 20.409

(a) Requirements for notifications and reports to individuals of exposure to radiation or radioactive material are specified in Section 19.13 of this chapter.

(b) When a licensee is required pursuant to Sections 20.405 or 20.408 to report to the Commission any exposure of an individual to radiation or radioactive material, the licensee shall also notify the individual. Such notice shall be transmitted at a time not later than the transmittal to the Commission, and shall comply with the provisions of Section 19.13(a) of this chapter.

EVALUATION OF COMPLIANCE

A copy of the letter generated in accordance with Section 20.408 is sent to the worker at the time the letter is submitted to the NRC.

CONCLUSION

Zion Station is in compliance with the intent of 10 CFR 20.409.

REFERENCES

Technical Services Nuclear Procedures

TSN-CR-D16 Instructions for Handling Computer Generated Dosimetry Termination Letters

10 CFR 50.34 CONTENTS OF APPLICATION: TECHNICAL INFORMATION

STATEMENT OF SECTION 50.34 - PARAGRAPH (b)

Final safety analysis report. Each application for a license to operate a facility shall include a final safety analysis report. The final safety analysis report shall include information that describes the facility, presents the design basis, and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole.

EVALUATION OF COMPLIANCE

The Zion Final Safety Analysis Report (FSAR) was docketed with the Atomic Energy Commission on November 25, 1970 pursuant to Section 50.34 of 10 CFR 50. This document and its numerous amendments were reviewed and approved by the Atomic Energy Commission in its Safety Evaluation Report, dated October 6, 1972. Subsequently, operating licenses were issued for Zion Unit 1 on April 6, 1973 and for Zion Unit 2 on November 14, 1973.

STATEMENT OF SECTION 50.34 - PARAGRAPH (c)

Physical security plan. Each application for a license to operate a production or utilization facility shall include a physical security plan. The plan shall consist of two parts. Part I shall address vital equipment, vital areas, and isolation zones, and shall demonstrate how the applicant plans to comply with the requirements of Part 73 of this chapter, if applicable, at the proposed facility. Part II shall list tests, inspections, and other means to be used to demonstrate compliance with such requirements, if applicable.

EVALUATION OF COMPLIANCE

The physical security plan for Zion Station was submitted to the Nuclear Regulatory Commission (NRC) in May 1977 with the most recent amendment submitted April 21, 1980. This plan has been approved by the NRC.

STATEMENT OF SECTION 50.34 - PARAGRAPH (d)

Safeguard contingency plan. Each application for a license to operate a production or utilization facility that shall be subject to Sections 73.50, 73.55, or 73.60 of this chapter shall include a licensee safeguards contingency plan in accordance with the criteria set forth in Appendix C to 10 CFR Part 73. The safeguards contingency plan shall include plans for dealing with threats, thefts, and industrial sabotage, as defined in Part 73 of this chapter, relating to the special nuclear material and nuclear facilities licensed under this chapter and in the applicant's possession and control. Each application for such a license shall include the first four categories of information contained in the applicant's safeguards contingency plan. (The first four categories of information, as set forth in Appendix C to 10 CFR Part 73, are Background, Generic Planning Base, Licensee Planning Base, and Responsibility Matrix. The fifth category of information, Procedures, does not have to be submitted for approval.)

EVALUATION OF COMPLIANCE

The safeguards contingency plan for Zion Station was submitted to the NRC on April 21, 1980. This plan is presently undergoing NRC review.

CONCLUSION

Zion has addressed and complies with the requirements of 10 CFR 50.34 with the exception that the safeguards contingency plan is still undergoing NRC staff review.

REFERENCES

1. B. Lee, CECo, letter to P. A. Morris, NRC, December 1, 1970 (FSAR Docketing Letter)

2. A. Giambusso, NRC, letter to B. Lee, CECo, April 6, 1973 (Zion Unit 1 Operating License)

3. K. R. Goller, NRC, letter to B. Lee, November 14, 1973 (Zion Unit 2 Operating License)

4. B. Lee, CECo, letter to E. Case, NRC, May, 1977 (Physical Security Plan)

5. A. Schwencer, NRC, letter to D. L. Peoples, CECo February 27, 1980 (Request for Safeguards Contingency Plan)

6. D. L. Peoples, CECO, letter to H. R. Denton, NRC April 21, 1980 (Safeguards Contingency Plan)

10 CFR 50.34a - DESIGN OBJECTIVES FOR EQUIPMENT TO CONTROL RELEASES OF RADIOACTIVE MATERIAL IN EFFLUENTS - NUCLEAR POWER REACTORS

STATEMENT SECTION 50.34a - PARAGRAPH (a)

An application for a permit to construct a nuclear power reactor shall include a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences. In the case of an application filed on or after January 2, 1971, the application shall also identify the design objectives, and the means to be employed, for keeping levels of radioactive material in effluents to unrestricted areas as low as is reasonably achievable. The term "as low as is reasonably achievable" as used in this part means as low as is reasonably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconmic considerations and in relation to the utilization of atomic energy in the public interest. The guides set out in Appendix I provide numerical guidance on design objectives for light water-cooled nuclear power reactors to meet the requirement that radioactive material in effluents released to unrestricted areas be kept as low as is reasonably achievable. These numerical guides for design objectives and limiting conditions for operation are not to be construed as radiation protection standards.

EVALUATION OF COMPLIANCE

The Zion construction permit application was filed on July 12, 1967. Zion compliance to Appendix I is documented in the Report, "Information Relevant to Keeping Levels of Radioactivity in Effluents to Unrestricted Areas As Low As Is Reasonably Achievable," Zion Station Units 1 & 2, June 4, 1976, and Amendment 1, November 12, 1976 (Appendix I Report).

STATEMENT OF SECTION 50.34 - PARAGRAPH (b)

Each application for a permit to construct a nuclear power reactor shall include:

1. A description of the preliminary design of equipment to be installed pursuant to Paragraph (a) of this section:

2. An estimate of:

   (i) The quantity of each of the principal radionuclides expected to be released annually to unrestricted areas in liquid effluents produced during normal reactor operations; and

(ii) The quantity of each of the principal radionuclides of the gases, halides, and particulates expected to be released annually to unrestricted areas in gaseous effluents produced during normal operations.

3. A general description of the provisions for packaging, storage, and shipment offsite of solid waste containing radioactive materials resulting from treatment of gaseous and liquid effluents and from other sources.

EVALUATION OF COMPLIANCE

1. Appendix I Report, Section 2.1

A description of the design of equipment installed at Zion is included in the Appendix I Report, Section 2.1.

2. Appendix I Report, Tables 1.1-1 and 1.1-5

The estimated annual releases of principal radionuclides in gaseous and liquid effuents are presented in Tables 1.1-1 and 1.1-5 of the Appendix I Report, respectively.

3. Appendix I Report, page 2.1-3

The following agreement concerning the description of the solid waste processing system was reached and is documented in the Appendix I Report, "In a meeting held at the NRC Region III offices on Tuesday, March 30, 1976, Commonwealth Edison Company was advised by NRC representatives that this section (i.e., Solid Radwaste Processing System) is not required for the Appendix I evaluation."

STATEMENT OF SECTION 50.34a - PARAGRAPH (c)

Each application for a license to operate a nuclear power reactor shall include (1) a description of the equipment and procedures for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems, pursuant to paragraph (a) of this section; and (2) a revised estimate of the information required in paragraph (b)(2) of this section if the expected releases and exposures differ significantly from the estimates submitted in the application for a construction permit.

EVALUATION OF COMPLIANCE

1. Appendix I Report, Zion Station, 1976

The information required in the paragraph concerning the description of equipment is contained in the Appendix I Report as discussed in the Evaluation of Compliance for Paragraphs (a) and (b).

2. Letter from D. L. Ziemann to H. R. Denton, dated February 19, 1976

The procedures for control of gaseous and liquid effluents are contained in the proposed Zion Appendix I Technical Specifications, Offsite Dose Calculation Manual (ODCM), and Environmental Monitoring Program which were submitted to the NRC on February 16, 1979 (Reference 5). NRC review of these items is not yet complete.

CONCLUSION

Zion Station has addressed and complies with the requirements of 10 CFR 50.34a as documented in the Appendix I Report. The proposed Technical Specifications and the ODCM are still undergoing NRC review.

REFERENCES

1. Appendix I Report, "Information Relevant to Keeping Levels of Radioactivity in Effluents to Unrestricted Areas As Low As Reasonably Achievable," Zion Station Units 1 & 2, June 4, 1976 and Amendment 1, November 12, 1976.

2. Radiological Safety Technical Specifications for Zion Station Units 1&2, Zion, Illinois.

   Sections 3.16 and 4.16 Environmental Radiological Monitoring Program

   Section 6.6.3a Unique Reporting Requirements - Semiannual Effluent Release Report

   Section 6.6.3b Unique Reporting Requirements - Environmental Radiological Monitoring

3. A. Schwencer, NRC, letter to R. L. Bolger, CECo, January 26, 1977.

4. R. L. Bolger, CECo, letter to A. Schwencer, April 26, 1977.

5. D. L. Ziemann, NRC, letter to R. L. Bolger, CECo, February 19, 1976.

6. C. Reed, CECo, letter to H. R. Denton, NRC, February 16, 1979.

10 CFR 50.36 - TECHNICAL SPECIFICATIONS

STATEMENT OF SECTION 50.36 - PARAGRAPHS (a) AND (b)

(a) Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section. A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications.

(b) Each license authorizing operation of a production or utilization facility of a type described in Section 50.21 or Section 50.22 will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to Section 50.34. The Commission may include such additional technical specifications as the Commission finds appropriate.

EVALUATION OF COMPLIANCE

The Zion Units 1 & 2 applications for operating licenses were approved in April and November 1973, respectively. These licenses incorporated the technical specifications of requirements of 10 CFR 50.36 - Paragraphs (a) and (b) above.

STATEMENT OF SECTION 50.36 - PARAGRAPH (c)

Technical Specifications will include items in the following categories:

(1) Safety limits, limiting safety system settings, and limiting control settings;

(2) Limiting conditions for operation;

(3) Surveillance requirements;

(4) Design Features; and

(5) Administrative controls.

EVALUATION OF COMPLIANCE

All of the above listed categories are contained in the current Zion Station Units 1 & 2 technical specifications. The technical specifications are a "living" document, having been amended over 50 times since the units' initial operation in 1973. The amendments were and will continue to be necessary to maintain the document current with NRC requirements, as well as with plant modifications for improved operations.

CONCLUSION

The technical specifications for Zion Units 1 & 2 meet the requirements of 10 CFR 50.36.

REFERENCES

Facility Operating License Nos. DPR-39 and DPR-48, Zion Nuclear Power Station Unit 1 and Unit 2, Appendix A - Radiological Safety Technical Specification.

10 CFR 50.36a - TECHNICAL SPECIFICATIONS ON EFFLUENTS FROM NUCLEAR POWER REACTORS

STATEMENT OF SECTION 50.36a - PARAGRAPHS (a) and (b)

(a) In order to keep releases of radioactive materials to unrestricted areas during normal reactor operations, including expected operational occurrences, as low as is reasonably achievable, each license authorizing operation of a nuclear power reactor will include technical specifications that, in addition to requiring compliance with applicable provisions of Section 20.106 of this chapter, require:

1. The operating procedures developed pursuant to Section 50.34a(c) for the control of effluents be established and followed and that equipment installed in the radioactive waste system pursuant to Section 50.34a(a) be maintained and used.

2. The submission of a report to the appropriate NRC Regional Office shown in Appendix D of Part 20 of this chapter within sixty (60) days after January 1 and July 1 of each year specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and in gaseous effluents during the previous six (6) months of operation, and such other information as may be required by the Commission to estimate maximum potential annual radiation doses to the public resulting from effluent releases. Copies of such report shall be sent to the Director of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. If quantities of radioactive materials released during the reporting period are significantly above design objectives, the report shall cover this specifically. On the basis of such report and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

(b) In establishing and implementing the operating procedures described in paragraph (a) of this section, the licensee shall be guided by the following considerations: Experience with the design, construction and operation of nuclear power reactors indicates that compliance with the technical specifications described in this section will keep average annual releases of radioactive material in effluents at small percentages of the limits specified in Section 20.106 of this chapter and in the operating license. At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power even under unusual operating conditions which may temporarily result in releases higher than such small percentages, but still within the limits specified in Section 20.106 of this chapter and the operating license. It is expected that in using this operational flexibility under unusual operating conditions, the licensee will exert his best efforts to keep levels of radioactive material in effluents as low as is reasonably achievable. The guides set out in Appendix I provide numerical guidance on limiting conditions for operation for light water-cooled nuclear power reactors to meet the requirement that radioactive materials in effluents released to unrestricted areas be kept as low as is reasonably achievable.

EVALUATION OF COMPLIANCE

1. Letter from D. L. Ziemann to R. L. Bolger, February 19, 1976

The operating procedures pursuant to paragraph 50.34a(c) are contained in the proposed Zion Appendix I Technical Specifications, the ODCM and Environmental Monitoring Program which were submitted to the NRC on February 16, 1979 (Reference 5). NRC review of these items is not yet complete.

2. Zion Station Technical Specifications, Subsection 6.6.3a

The Zion requirements for reporting semiannual releases in Subsection 6.6.3a of the Zion Technical Specifications are in conformance with the above requirement.

CONCLUSIONS

The proposed Zion Appendix I Technical Specifications, ODCM, and Environmental Monitoring Program have been submitted to the NRC. The NRC review of these submittals is not yet complete.

REFERENCES

1. "Information Relevant To Keeping Levels of Radioactivity in Effluents to Unrestricted Areas As Low As Reasonably Achievable," Zion Station Units 1&2, June 4, 1976 and Amendment 1, November 12, 1976. (Appendix I Report)

2. Radiological Safety Technical Specifications for Zion Station Units 1 & 2, Zion, Illinois.

   Sections 3.16 and 4.16 Environmental Radiological Monitoring Program

   Subsection 6.6.3a Unique Reporting Requirements - Semiannual Effluent Release Report

Subsection 6.6.3b Unique Reporting Requirements - Environmental Radiological Monitoring

3. A Schwencer, NRC, letter to R. L. Bolger, CECo, January 26, 1977.

4. R. L. Bolger, CECo, letter to A. Schwencer, NRC, April 26, 1977.

5. D. L. Ziemann, NRC, letter to R. L. Bolger, CECo, February 19, 1976.

6. C. Reed, CECo, letter to H. R. Denton, NRC, February 16, 1979.

10 CFR 50.44 STANDARDS FOR COMBUSTIBLE GAS CONTROL SYSTEM IN LIGHT-WATER COOLED POWER REACTORS

STATEMENT OF SECTION 50.44 - PARAGRAPH (a)

Each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding, shall, as provided in paragraphs (b) through (d) of this section, include means for control of hydrogen gas that may be generated, following a postulated loss-of-coolant accident (LOCA), by (1) metal-water reaction involving the fuel cladding and the reactor coolant, (2) radiolytic decomposition of the reactor coolant, and (3) corrosion of metals.

EVALUATION OF COMPLIANCE

Two modes of hydrogen gas control are available at Zion Station:

a. Hydrogen recombiners (FSAR, Section 6.8), and

b. The containment purge system (FSAR, Section 14.3.6).

STATEMENT OF SECTION 50.44 - PARAGRAPH (b)

Each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding shall be provided with the capability for (1) measuring the hydrogen concentration in the containment, (2) insuring a mixed atmosphere in the containment, and (3) controlling combustible gas concentrations in the containment following a postulated LOCA.

EVALUATION OF COMPLIANCE

1. FSAR, Section 11.3.3.3

Hydrogen concentration is measured by taking bottled air samples of the containment atmosphere, following an accident involving loss of reactor coolant, by using the containment area sampling system.

2. FSAR, Question Q14.17

Operation of the containment fan coolers creates a mixing of the containment atmosphere.

3. FSAR, Section 6.8 and 14.3.6

Two methods are available to control combustible gas concentration in containment following a postulated LOCA. These methods are use of the hydrogen recombiner and use of the containment purge system.

STATEMENT OF SECTION 50.44 - PARAGRAPH (c)

For each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding, it shall be shown that during the time period following a postulated LOCA but prior to effective operation of the combustible gas control system, either: (1) An uncontrolled hydrogen-oxygen recombination would not take place in the containment; or (2) the plant could withstand the consequences of uncontrolled hydrogen-oxygen recombination without loss of safety function. If neither of these conditions can be shown, the containment shall be provided with an inerted atmosphere or an oxygen deficient condition in order to provide protection against hydrogen burning and explosions during this time period.

EVALUATION OF COMPLIANCE

Analysis shows that hydrogen concentration inside containment following a loss-of-coolant accident will not reach the lower flammability limit for 37 days. Containment purging can begin 30 days after an accident without exceeding 10 CFR 100 limits (FSAR, Question Q14.2).

The hydrogen recombiner can also be installed and brought on line in sufficient time to control hydrogen concentration in the containment below the lower flammability limit.

STATEMENT OF SECTION 50.44 - PARAGRAPH (d)

(1) For facilities that are in compliance with Section 50.46(b), the amount of hydrogen contributed by core metal-water reaction (percentage of fuel cladding that reacts with water), as a result of degradation, but not total failure, of emergency core cooling functioning shall be assumed either to be five times the total amount of hydrogen calculated in demonstrating compliance with Section 50.46(b)(3), or to be the amount that would result from reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) to a depth of 0.0023 inch (0.0058 mm), whichever amount is greater. A time period of 2 minutes shall be used as the interval after the postulated LOCA over which the metal-water reaction occurs. (2) For facilities as to which no evaluation of compliance in accordance with Section 50.46(b) has been submitted and evaluated, the amounts of hydrogen so contributed shall be assumed to be that amount resulting from the reactor of 5 percent of the mass of metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume.

EVALUATION OF COMPLIANCE

The analysis performed to determine hydrogen generation at Zion was done in accordance with Safety Guide 7. This guide assumed that the extent of metal-water reaction (percentage of fuel cladding that reacts with water) was 5% (FSAR, Question 14.2).

STATEMENT OF SECTION 50.44 - PARAGRAPH (e)

For facilities whose notice of hearing on the application for a construction permit was published on or after November 5, 1970...

EVALUTION OF COMPLIANCE

Notice of heating on the application for a construction permit for Zion Units 1 & 2 was published prior to November 5, 1970.

STATEMENT OF SECTION 50.44 - PARAGRAPH (f)

For facilities with respect to which the notice of hearing on the application for a construction permit was published between December 22, 1968, and November 5, 1970...

EVALUATION OF COMPLIANCE

Notice of hearing on the application for a construction permit for Zion Units 1 & 2 was published prior to December 22, 1968.

STATEMENT OF SECTION 50.44 - PARAGRAPH (g)

For facilities with respect to which the notice of hearing on the application for a construction permit was published on or before December 22, 1968, if the combined radiation dose at the low population zone outer boundary from purging (and repressurization if a repressurization system is provided) and the postulated LOCA calculated in accordance with Section 100.11(a)(2) of this chapter is less than 25 rem to the whole body and less than 300 rem to the thyroid, only a purging system is necessary, provided that the purging system and any filtration system associated with it are designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. If a purge syrtem is used as part of the repressurization system, it shall be designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. The containment shall not be repressurized beyond 50 percent of the containment design pressure.

EVALUATION OF COMPLIANCE

1. FSAR, Question 1.10

The containment purge system is designed to comply with the intent of Criteria 41, 42, and 43 of Appendix A to 10 CFR 50 as discussed in the evaluation of compliance for each of these criteria.

2. FSAR, Question 14.2, and SER, Table 15.1

The containment doses due to purging at the low population zone outer boundary is less than 25 rem to the whole body and 300 rem to the thyroid as shown in the response to FSAR Question 14.2 and Table 15.1 of the Zion Safety Evaluation Report.

3. FSAR, Section 6.8

Zion Units 1 & 2 are also provided with hydrogen recombiners.

CONCLUSION

The combustible gas control system at Zion meets the requirements of 10 CFR 50.44.

REFERENCES

FSAR Sections

6.8 Recombiner

11.3.3.3 Containment Area Sampling System

14.3.6 Controlled Containment Venting After a LOCA

FSAR Questions

Questions 1.10, 14.2, 14.16 and 14.17

Safety Guides

Safety Guide 7 Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident (dated March 10, 1971).

SER

Section 15.0 Accident Analyses

10 CFR 50.46 - ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT WATER NUCLEAR POWER REACTORS

Compliance with 10 CFR 50.46 is documented in Amendment No. 53 to Facility Operating License No. DPR-39 and Amendment No. 50 to Facility Operating License No. DPR-48, for Zion Station Units 1 and 2. This amendment meets 10 CFR 50.46 criteria and was approved by the NRC as indicated in the conclusion of the Safety Evaluation Report which states that:

"Based on the review of the submitted documents, we conclude that the results of the LOCA analysis performed with F_Q=1.93 are conservative relative to the 10 CFR 50.46 criteria. We consider the resulting changes to the Technical Specifications acceptable for operating Units 1 and 2 with a maxium of 1 percent of the steam generator tubes plugged."

In addition to the preceding, a LOCA analysis has also been performed with F_Q = 2.20 and was submitted to the NRC Staff for approval on October 22, 1979. This analysis confirmed that the Zion units could be operated with the higher F_Q = 2.20, while still being conservative relative to the 10 CFR 50.46 criteria.

10 CFR 50.54 - CONDITIONS OF LICENSES

STATEMENT OF SECTION 50.54 - PARAGRAPHS (a) THROUGH (h)

Whether stated therein or not, the following shall be deemed conditions in every license issued:

(a) [Deleted, effective March 9, 1967 (32 F. R. 2562).]

(b) No right to the special nuclear material shall be conferred by the license except as may be defined by the license.

(c) Neither the license, nor any right thereunder, nor any right to utilize or produce special nuclear material shall be transferred, assigned, or disposed of in any manner, either voluntarily or involuntarily, directly or indirectly, through transfer of control of the license to any person, unless the Commission shall, after securing full information, find that the transfer is in accordance with the provisions of the Act and give its consent in writing.

(d) The license shall be subject to suspension and to the rights of recapture of the material or control of the facility reserved to the Commission under Section 108 of the Act in a state of war or national emergency declared by Congress.

(e) The license shall be subject to revocation, suspension, modification, or amendment for cause as provided in the Act and regulations, in accordance with the procedures provided by the Act and regulations.

(f) The licensee will at any time before expiration of the license, upon request of the Commission submit written statements, signed under oath or affirmation, to enable the Commission to determine whether or not the license shall be modified, suspended or revoked.

(g) The issuance or existence of the license shall not be deemed to waive, or relieve the license from compliance with, the antitrust laws, as specified in subsection 105a of the Act. In the event that the licensee should be found by a court of competent jurisdiction to have violated any provision of such antitrust laws in the conduct of the licensed activity, the Commission may suspend or revoke the license or take such other action with respect to it as shall be deemed necessary.

(h) The license shall be subject to the provisions of the Act now or hereafter in effect and to all rules, regulations, and orders of the Commission. The terms and conditions of the license shall be subject to amendment, revision, or modification, by reason of amendments of the Act or by reason of rules, regulations, and orders issued in accordance with the terms of the Act.

EVALUATION OF COMPLIANCE

Commonwealth Edison Company acknowledges the above listed conditions and complies with them.

STATEMENT OF SECTION 50.54 - PARAGRAPH (i)

Except as provided in Section 55.9 of this chapter, the licensee shall not permit the manipulation of the controls of any facility by anyone who is not a licensed operator or senior operator as provided in Part 55 of this chapter.

EVALUATION OF COMPLIANCE

A trained licensed Nuclear Station Operator is assigned to responsiblity for the operation of each operating reactor each shift (FSAR, Section 12.1.1, page 12.1-1).

STATEMENT OF SECTION 50.54 - PARAGRAPH (i-1)

Within three (3) months after issuance of an operating license, the licensee shall have in effect an operator requalification program which shall, as a minimum, meet the requirements of Appendix A of Part 55 of this Chapter. Notwithstanding the provisions of Section 50.59 the licensee shall not except as specifically authorized by the Commission, make a change in an approved operator qualification program by which the scope, time allotted for the program or frequency in conducting different parts of the program is decreased. Holders of operating licenses in effect on September 17, 1973, shall implement an operator requalification program which, as a minimum, meets the requirements of Appendix A of Part 55 of this chapter which was submitted for approval by the Atomic Energy Commission.

EVALUATION OF COMPLIANCE

Retraining and replacement training of Station personnel shall in accordance with ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971.

Retraining shall be conducted at intervals not exceeding two years (Zion Station Radiological Safety Technical Specifications, page 300). However, in accordance with Item B.8 of Zion Confirmatory Order of February 29, 1980, retraining is currently conducted on an annual basis.

STATEMENT OF SECTION 50.54 - PARAGRAPHS (j) AND (k)

(j) Apparatus and mechanisms other than controls, the operation of which may affect the reactivity or power level of a reactor shall be manipulated only with the knowledge and consent of an operator or senior operator licensed pursuant to Part 55 of this chapter present at the controls.

(k) An operator or senior operator licensed pursuant to Part 55 of this chapter shall be present at the controls at all times during the operation of the facility.

EVALUATION OF COMPLIANCE

A trained licensed Nuclear Station Operator is assigned to responsiblity for the operation of each operating reactor each shift (FSAR, Section 12.1.1, page 12.1-1).

STATEMENT OF SECTION 50.54 - PARAGRAPH (l)

The licensee shall designate individuals to be responsible for directing the licensed activities of licensed operators. These individuals shall be licensed as senior operators pursuant to Part 55 of this chapter.

EVALUATION OF COMPLIANCE

Figure 6.1-2 of the Zion Radiological Safety Technical Specifications depicts the Zion Station organization which shows that the Shift Foreman and Shift Engineer who must have Senior Operator licenses are responsible for directing the licensed activities of licensed operators.

STATEMENT OF SECTION 50.54 - PARAGRAPH (m)

A senior operator licensed pursuant to Part 55 of this chapter shall be present at the facility or readily available on call at all times during its operation, and shall be present at the facility during initial start-up and approach to power, recovery from an unplanned or unscheduled shut-down or significant reduction in power, and refueling, or as otherwise prescribed in the facility license.

EVALUATION OF COMPLIANCE

Figure 6.1-3 of the Zion Radiological Safety Technical Specifications depicts the Zion Shift Manning Chart which shows that a senior operator is assigned to each shift.

STATEMENT OF SECTION 50.54 - PARAGRAPH (n)

The licensee shall not, except as authorized pursuant to a construction permit, make any alteration in the facility constituting a change from the technical specifications previously incorporated in a license or construction permit pursuant to Section 50.36.

EVALUATION OF COMPLIANCE

Commonwealth Edison Company acknowledges and complies with this requirement. All proposed changes pursuant to 10 CFR 50.59 are reviewed to verify that such actions do not constitute an unreviewed safety question.

STATEMENT OF SECTION 50.54 - PARAGRAPH (o)

Primary reactor containment for water cooled power reactors shall be subject to the requirements set forth in Appendix J.

EVALUATION OF COMPLIANCE

Zion Station complies with the intent of 10 CFR 50 Appendix J as discussed in the Appendix J analysis.

STATEMENT OF SECTION 50.54 - PARAGRAPH (p)

The licensee shall make no change which would decrease the effectiveness of a security plan prepared pursuant to Section 50.34(c) or Part 73 of this chapter without the prior approval of the Commission. A licensee desiring to make such a change shall submit an application for an amendment to his license pursuant to Section 50.90. The licensee shall maintain records of changes to the plan made without prior Commission approval for a period of two years from the date of the change, and shall furnish to the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the appropriate NRC Regional Office specified in Appendix D of Part 20 of this chpater, a report containing a description of each change within two months after the change is made.

EVALUATION OF COMPLIANCE

Commonwealth Edison Company acknowledges and complies with this requirement. Changes to the security plan are maintained in the Commonwealth Edison Company files.

CONCLUSIONS

Zion complies with the requirements of 10 CFR 50.54

REFERENCES

1. Zion Station Final Safety Analysis Report, Section 12.1.1, Organization.

2. Zion Station Radiological Safety Technical Specifications, Section 6.0, Administrative Controls, pages 300-333.

3. H. R. Denton, NRC, letter to D. L. Peoples, CECo, February 29, 1980 (Confirmatory Order).

SECTION 50.55a - CODES & STANDARDS

STATEMENT OF SECTION 50.55a - PARAGRAPH (a)(1)

Structures, systems, and components shall be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

EVALUATION OF COMPLIANCE

1. FSAR, page 1.2-2

Zion structures, systems, and components are classified in the FSAR.

a. Seismic Class I - Failure would cause or increase the severity of a LOCA; vital to safe shutdown and isolation,

b. Seismic Class II - Function in direct support of reactor operation,

c. Seismic Class III - Neither Class I or II.

2. FSAR, Question 1.5, page Q1.5-2.3

As part of the Zion QA program, a classification is established... to provide control over activities affecting the quality of the safety-related Seismic Class I structures, systems, and components...

3. FSAR, Appendix 1 to Question 1.5

Seismic Class I structures, systems and components are identified in Appendix 1 to FSAR Question 1.5.

Zion structures, systems and components are designed to the codes and standards listed in the FSAR Tables 4.1.11 (Reactor Coolant System), Chapter 5.0 (Containment Structure), Table 6.2-1 (ECCS Systems), Table 9.2-1 (CVCS), Table 9.3-1 (Component Cooling System), Table 9.4-1 (RHRS), Table 9.5-1 (Spent Fuel Pit Cooling System), and Table 9.8-1 (Sampling System).

STATEMENT OF SECTION 50.55a - PARAGRAPH (a)(2)

As a minimum, the systems and components of boiling and pressurized water-cooled nuclear power reactors specified in paragraphs (c), (d), (e), (f), (g), and (i) of this section shall meet the requirements described in those paragraphs, except that the American Society of Mechanical Engineers (hereinafter referred to as ASME) Code N-symbol need not be applied, and the protection systems of nuclear power reactors of all types shall meet the requirements described in paragraph (h) of this section, except as authorized by the Commission or the Atomic Energy Commission upon demonstration by the applicant for or holder of a construction permit that:

(i) Design, fabrication, installation, testing of the specified system or component, is to the maximum extent practical, in accordance with generally recognized codes and standards, and compliance with the requirements described in paragraphs (c) through (i) of this section or portions thereof would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety; or

(ii) Proposed alternatives to the described requirements or portions thereof will provide an acceptable level of quality and safety. For example, the use of inspection or survey systems other then those required by the specified ASME Codes and Addenda may be authorized under this subparagraph provided that an acceptable level of quality and safety in design, fabrication, installation, and testing is achieved.

EVAULATION OF COMPLIANCE

An evaluation of compliance is provided for each of the 10 CFR 50.55a paragraphs (c), (d), (e), (f), (h), and (i) referenced in paragraph (a)(2) above.

STATEMENT OF SECTION 50.55a - PARAGRAPH (b)

(1) As used in this section, references to Section III of the ASME Boiler and Pressure Vessel Code refer to Section III, Division 1, and include editions through the 1977 Edition and addenda through the Summer 1978 Addenda.

(2) As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Section XI, Division 1 and include editions through the 1977 Edition and addenda through the Summer 1978 Addenda, subject to the following limitations and modifications:

(i) Applicability of specific editions and addenda. When applying the 1974 Edition, only the addenda through the Summer 1975 Addenda may be used. When applying the 1977 Edition, all of the addenda through the Summer 1978 Addenda must be used.

(ii) Pressure-retaining welds in ASME Code Class 1 piping (applies to Table IWB-2500 and IWB-2500-1 and Category B-J). If the facility's application for a construction permit was docketed prior to July 1, 1978, the extent of examination for Code Class 1 pipe welds may be determined by the requirements of Table IWB-2500 and Table IWB-2600, Category B-J of Section XI of the ASME Code in the 1974

Edition and addenda through the Summer 1975 Addenda or other requirements the Commission may adopt.

(iii) Steam generator tubing (modifies Article IWB-2000). If the technical specifications of a nuclear power plant include surveillance requirements for steam generators different than those in Article IWB-2000, the inservice inspection program for steam generator tubing shall be governed by the requirements in the technical specifications.

(iv) Pressure-retaining welds in ASME Code Class 2 piping (applies to Tables IWC-2520 or IWC-2520-1, Category C-F. (A) Appropriate Code Class 2 pipe welds in Residual Heat Removal Systems, Emergency Core Cooling Systems, and Containment Heat Removal Systems, shall be examined. The extent of examination for these systems shall be determined by the requirements of paragraph IWC-1220, Table IWC-2520 Category C-F and C-G, and paragraph IWC-2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda of Section XI of the ASME Code.

(B) For a nuclear power plant whose application for a construction permit is docketed prior to July 1, 1978, the extent of examination for Code Class 2 pipe welds may be determined by the requirements of paragraph, IWC-1220, Table IWC-2520 Category C-F and C-G and paragraph IWC-2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda of Section XI of the ASME Code or other requirements the Commission may adopt.

EVALUATION OF COMPLIANCE

Zion was designed to the ASME Code editions and addenda described in the evaluation of compliance for 10 CFR 50.55a - Paragraphs (c), (d), (e), (f), and (g) below.

1. References 1, 2, and 3

The currently submitted Zion ISI programs for the second 40-month period meet the requirement of Section XI of the ASME Code, 1974 edition, Summer 1975 Addenda.

2. Zion Station Radiological Safety Technical Specifications, Sections 3/4.3

Sections 3/4.3 of the Technical Specifications address Inservice Inspection, including ISI of the steam generators.

3. References 4, 5, 6, 7, and 12

The Zion Station ISI Program received NRC approval on August 8, 1980. Since only NRC interim approval was received prior to final approval for the Zion second 40-month period ISI programs, Zion Station performed inservice inspections and tests either to the requirements in the submitted program or to the requirements in the Technical Specifications whichever was more conservative.

STATEMENT OF SECTION 50.55a - PARAGRAPH (c)

Pressure vessels: (1) For construction permits issued before January 1, 1971, for reactors not licensed for operating pressure vessels which are part of the reactor coolant pressure boundary shall meet the requirements for Class A vessels set forth in Section III of the ASME Boiler and Pressure Vessel Code, applicable Code Cases, and Addenda in effect on the date of order of the vessel. The pressure vessels may meet the requirements set for in editions of this Code, applicable Code Cases, and Addenda which have become effective after the date of vessel order.

(2) For contruction permits issued on or after January 1, 1971, but before July 1, 1974, pressure vessels which are part of the reactor coolant pressure boundary shall meet the requirements for Class A or Class 1 vessels set forth in Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the pressure vessel: Provided, however, that if the pressure vessel is ordered more than 18 months prior to the date of issuance of the construction permit, compliance with the requirements for Class A or Class 1 vessels set forth in editions of Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect 18 months prior to the date of issuance of the construction permit is required. The pressure vessels may meet the requirements set forth in editions of this Code and Addenda which have become effective after the date of ve[illegible] order or after 18 months prior to the date of issuance of [illegible] construction permit.

(3) For construction permits issued on or after July 1, 1974, pressure vessels which are part of the reactor coolant pressure boundary shall meet the requirements for Class 1 components set forth in Section III of the ASME Boiler and Pressure Vessel Code: Provided, that the ASME Code provisions applied to the pressure vessels shall be no earlier than those of the Summer 1972 Addenda of the 1971 edition.

EVALUATION OF COMPLIANCE

The Zion construction permit was issued on December 26, 1968; therefore, 10 CFR 50.55a, Paragraph (c)(1) applies.

The Unit 1 and Unit 2 Reactor Vessels were designed to the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, 1965 Edition, through Summer 1966 addenda, Code Cases 1332-4, 1335-2, 1338-4, 1358-1, and 1359-1 (FSAR, Question 4.14).

STATEMENT OF SECTION 50.55a - PARAGRAPH (d)

Piping: (1) For construction permits issued before January 1, 1971, for reactors not licensed for operation, piping which is part of the reactor coolant pressure boundary shall meet the requirements set forth in:

> (i) The American Standard Code for Pressure Piping (ASA B31.1), Addenda, and applicable Code Cases or the Class I Section of the USA Standard Code for Pressure Piping (USAS B31.7) in effect on the date of order of the piping and
>
> (ii) The nondestructive examination and acceptance standards of ASA B31.1 Code Cases N7, N9, and N10, except that the acceptance standards of Class I piping of the USA Standard Code for Pressure Piping (USAS B31.7) may be applied.
>
> The piping may meet the requirements set forth in editions of ASA B31.1, USAS B31.1.0 and USAS B31.7 Addenda and Code Cases which become effective after the date of order of the piping.

(2) For construction permits issued on or after January 1, 1971 but before July 1, 1974, piping which is part of the reactor coolant pressure boundary shall meet the requirements for Class 1 piping set forth in the USA Standard Code for Pressure Piping (USAS B31.7) and Addenda in effect on the date of order of the piping and the requirements applicable to piping of articles 1 and 8 of editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the piping, or (ii) the requirements applicable to Class 1 piping of editions of Section III of the ASME Boiler and Pressure Vessel Code Addenda in effect on the date of the order of the piping: Provided, however, that if the piping is ordered more than 6 months prior to the date of issuance of the construction, permit compliance with the requirements for Class I or Class 1 piping set forth in editions of USAS B31.7 or Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect 6 months prior to the date of issuance of the construction permit is required. The piping may meet the requirements set forth in editions of these Codes and Addenda which have become effective after the date of piping order or after 6 months prior to the date of issuance of the construction permit.

(3) For construction permits issued on or after July 1, 1974, piping which is part of the reactor coolant pressure boundary shall meet the requirements for Class 1 components set forth in Section III of the ASME Boiler and Pressure Vessel Code. Provided, that the ASME Code provisions applied to the piping shall be no earlier than those of Winter 1972 Addenda of the 1971 edition.

EVALUATION OF COMPLIANCE

The Zion construction permit was issued on December 26, 1968, therefore, 10 CFR 50.55a paragraph (d)(1) applies.

Reactor coolant main loop piping and fitting were designed to USAS B31.1, 1955 Edition, Code Cases N7 and N10. Reactor coolant branch line piping was designed to USAS B31.1, Code Case N7 1967 Edition (FSAR Question 4.14).

STATEMENT OF SECTION 50.55a - PARAGRAPH (e)

Pumps: (1) For construction permits issued before January 1, 1971, for reactors not licensed for operation, pumps which are part of the reactor coolant pressure boundary shall meet___

(i) The requirements for Class I pumps set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power, Addenda, and Code Cases in effect on the date of order of the pumps, or

(ii) The nondestructive examination and acceptance standards set forth in ASA B31.1 Code Cases N7, N9, and N10, except that the acceptance standards for Class I pumps set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda in effect on the date of order of the pumps may be applied.

The pumps may meet the requirements set forth in editions of the Draft ASME Code for Pumps and Valves for Nuclear Power, Addenda, and Code Cases which become effective after the date of order of the pumps.

(2) For construction permits issued on or after January 1, 1971, but before July 1, 1974, pumps which are part of the reactor coolant pressure boundary shall meet the requirements for Class I pumps set forth in editions of (i) the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda in effect on the date of order of the pumps and the requirements applicable to pumps set forth in articles 1 and 8 of Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the pumps, or (ii) the requirements applicable to Class 1 pumps of editions of Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the pumps: Provided, however, that if the pumps are ordered more than 12 months prior to the date of issuance of the construction permit, compliance with the requirements for Class I pumps set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda and the requirements applicable to pumps set forth in articles 1 and 8 of editions of Section III of the ASME Boiler and Pressure Vessel Code and Addenda, or for Class 1 pumps of editions of Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect 12 months prior to the date of issuance of the construction permit is required. The pumps may

meet the requirement set forth in editions of these Codes or Addenda which have become effective after the date of pump order or after 12 months prior to the date of issuance of the construction permit.

(3) For construction permits issued on or after July 1, 1974, pumps which are part of the reactor coolant pressure boundary shall meet the requirements for Class 1 components set forth in Section III of the ASME Boiler and Pressure Vessel Code: Provided, that the ASME Code provisions applied to the pumps shall be no earlier than those of the Winter 1972 Addenda of the 1971 edition.

EVALUATION OF COMPLIANCE

The Zion Construction permit was issued on December 26, 1968, therefore, paragraph (e)(1) applies.

The reactor coolant pump casings were designed in accordance with ASME III, Article 4, 1968 edition (FSAR, Question 4.14).

STATEMENT OF SECTION 50.55a - PARAGRAPH (f)

Valves: (1) For construction permits issued before January 1, 1971, for reactors not licensed for operation, valves which are part of the reactor coolant pressure boundary shall meet the requirements set forth in

(i) The American Standard Code for Pressure Piping (ASA B31.1), Addenda, and applicable Code Cases, or the USA Standard Code for Pressure Piping (USAS B31.1.0), Addenda, and applicable Code Cases, in effect on the date of order of the valves or the Class I section of the Draft ASME Code for Pumps and Valves for Nuclear Power Addenda, and Code Cases in effect on the date of order of the valves or

(ii) The nondestructive examination and acceptance standards of ASA B31.1 Code Cases N2, N7, N9, and N10, except that the acceptance standards for Class 1 valves set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda in effect on the date of order of the valves may be applied.

The valves may meet the requirements set forth in editions of ASA B31.1, USAS B31.1.0, and the Draft ASME Code for Pumps and Valves for Nuclear Power, Addenda, and Code Cases which become effective after the date of order of the valves.

(2) For construction permits issued on or after January 1, 1971, but before July 1, 1974, valves which are part of the reactor coolant pressure boundary shall meet the requirements for Class I valves set forth in editions of (i) the Draft ASME Code for Pumps

and Valves for Nuclear Power and Addenda in effect on the date of order of the valves and the requirements applicable to valves set forth in articles 1 and 8 of editions of Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the valves, or (ii) the requirements applicable to Class 1 valves of section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the valve: Provided, however, that if the valves are order more than 12 months prior to the date of issuance of the construction permit, compliance with the requirements for Class 1 valves set forth in editions of the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda and the requirements applicable to valves set forth in articles 1 and 8 of editions of Section III of the ASME Boiler and Pressure Vessel Code and Addenda or for Class 1 valves of Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect 12 months prior to the date of issuance of the construction, permit is required. The valves may meet the requirements set forth in editions of these codes or Addenda which have become effective after the date of valve order or after 12 months prior to the date of issuance of the construction permit.

(3) For construction permits issued on or after July 1, 1974, valves which are part of the reactor coolant pressure boundary shall meet the requirements set forth in Section III of the ASME Boiler and Pressure Vessel Codes, provided, that the ASME Code provisions applied to the valves shall be no earlier than those of the Winter 1972 Addenda of the 1971 edition.

EVALUATION OF COMPLIANCE

The Zion construction permit was issued on December 26, 1968, therefore, paragraph (f)(1) applies.

The Loop Isolation valves are designed to the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, 1965 edition through Winter 1968 Addenda. Other valves were designed to ANSI B16.5 and MSS-SP-66 (FSAR, Question 4.14).

STATEMENT OF SECTION 50.55a - PARAGRAPH (g)(1) THROUGH (g)(3)

(1) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued prior to January 1, 1971, components (including supports) shall meet the requirements of paragraphs (g)(4) and (g)(5) of this section to the extent practical. Components which are part of the reactor coolant pressure boundary and their supports shall meet the requirements applicable to components which are classified as ASME Code Class 1. Other safety-related pressure vessels, piping, pumps and valves shall meet the requirements applicable to components which are classified as ASME Code Class 2 or Class 3.

(2) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after

January 1, 1971, but before July 1, 1974, components (including supports) which are classified as ASME Code Class 1 and Class 2 shall be designed and be provided with access to enable the performance of (i) inservice examination of such components (including supports) and (ii) tests for operational readiness of pumps and valves, and shall meet the preservice examination requirements set forth in editions of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda in effect 6 months prior to the date of issuance of the construction permit. The components (including supports) may meet the requirements set forth in subsequent editions of this code and addenda which are incorporated by reference in paragraph (b) of this section, subject, to the limitations and modifications listed therein.

(3) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after July 1, 1974:

(i) Components which are classified as ASME Code Class 1 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular component in accordance with paragraphs (c), (d), (e), or (f) of the section.

(ii) Components which are classified as ASME Code Class 2 and Class 3 and supports for components which are classified as ASME Code Class 1, Class 2, and Class 3 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular component.

(iii) Pumps and valves which are classified as ASME Code Class 1 shall be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular pump or valve in accordance with paragraphs (c) and (f) of this section or the Summer 1973 Addenda, whichever is later.

(iv) Pumps and valves which are classified as ASME Code Class 2 and Class 3 shall be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in Section XI of editions of the Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(v) All components (including supports) may meet the requirements set forth in subsequent editions of codes and addenda or portions thereof which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed therein.

EVALUATION OF COMPLIANCE

The Zion construction permit was issued on December 26, 1968, therefore, paragraph (g)(1) is applicable.

Zion safety-related components have been classified as ASME Code Class 1, 2 or 3 (References 1, 2, and 3).

STATEMENT OF SECTION 50.55a - PARAGRAPH (g)(4)

(4) Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which are classified as ASME Code Class 1, Class 2 and Class 3 shall meet the requirements except design and access provisions and preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda that become effective subsequent to editions specified in paragraphs (g)(2) and (g)(3) of this section and are incorporated by reference in paragraph (b) of this section to the extent practical within the limitations of design, geometry and materials of construction of the components.

(i) Inservice examinations of components, inservice tests to verify operational readiness of pumps and valves whose function is required for safety and system pressure tests, conducted during the initial 120-month inspection interval shall comply with the requirements in the latest edition and addenda of the code incorporated by reference in paragraph (b) of this section of the operating license, subject to the limitations and modifications listed in paragraph (b) of this section.

(ii) Inservice examinations of components, inservice tests to verify operational readiness of pumps and valves whose function is required for safety and system pressure tests, conducted during successive 120-month inspection intervals shall comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months prior to the start of the 120-month inspection interval, subject to the limitations and modifications listed in paragraph (b) of this section.

(iii) For a facility whose operating license was issued prior to March 1, 1976, the provisions of paragraph (g)(4) of this section are effective after September 1, 1976, at the start of the next one-third of a 120-month inspection interval. During that third of an inspection interval and the remainder of the inspection interval, the inservice examinations of components, tests to verify operational readiness of pumps and valves whose function is required for safety and system pressure tests, for such facilities shall comply with the requirements in the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section on the date 12-month prior to the start of that third of an inspection interval, subject to the limitations and modifications listed in paragraph (b) of this section.

(iv) Inservice examinations of components, tests of pumps and valves, and system pressure tests, may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section and subject to Commission approval. Portions of editions of addenda may be used provided that all related requirements of the respective editions or addenda are met.

EVALUATION OF COMPLIANCE

1. FSAR, Question 4.10

The baseline and preliminary inservice inspection program for Zion complies with ASME Code Section XI including the Winter 1970 Addenda, to the extent that the design of the plant, state of nondestructive testing technology and access to areas to be inspected will allow.

2. References 1, 2, 3, and 8

On May 27, 1977, the proposed Technical Specification amendments to incorporate the requirements of 10 CFR 50.55a for inservice inspection and pump and valve inservice testing were submitted to the NRC. The Zion Unit 1 ISI program and requests for exemptions to comply with 10 CFR 50.55a were submitted at that time also for the second 40-month period of the first 10-year interval. The Zion Unit 2 program was submitted on June 29, 1977 (Reference 2). Revised Inservice Inspection Programs for the second 40-month period of the first 10-year interval were submitted on June 28, 1979 (Reference 8).

3. References 4, 5, and 12

On December 7, 1977, the NRC on the basis of their preliminary review granted relief on an interim basis from those inservice inspection and testing requirements of the ASME Code requested for Zion Units 1 & 2 in References 4 and 5. NRC review of the

Zion ISI program was completed on approval granted on August 8, 1980 (Reference 12).

4. References 1 and 2

The Zion ISI inspection program for the second 40-month period of the first 10-year interval meets the requirements of Section XI of the ASME Code, 1974 Edition, Summer 1975 Addenda. This 40/20 month cycle started on April 30, 1977, and will be completed on August 31, 1980 for Unit 1 and started January 17, 1978 for Unit 2.

5. References 6, 7, 8, 9, 10, 11, and 12

Additional exemptions were requested on December 13, 1977; March 15, 1978; June 28 and October 1, 1979; and March 4, 1980. The October 1, 1979 request for exemption was granted on October 26, 1979. The remaining requests were granted on August 8, 1980.

STATEMENT OF SECTION 50.55a - PARAGRAPH (g)(5) AND (g)(6)

(5)(i) The inservice inspection program for a boiling or pressurized water-cooled nuclear power facility shall be revised by the licensee, as necessary to meet the requirements of paragraph (g)(4) of this section.

(ii) If a revised inservice inspection program for a facility conflicts with the technical specification for the facility, the licensee shall apply to the Commission for amendment of the technical specifications to conform the technical specification to the revised program. This application shall be submitted at least 6 months before the start of the period during which the provisions become applicable as determined by paragraph (g)(4) of this section.

(iii) If a licensee has determined that conformance with certain code requirements is impractical for his facility, the licensee shall notify the Commission and submit information to support his determinations.

(iv) Where an examination or test requirement by the code or addenda is determined to be impractical by the licensee and is not included in the revised inservice inspection program as permitted by paragraph (g)(4) of this section, the basis for this determination shall be demonstrated to the satisfaction of the Commission not later than 12 months after the expiration of the initial 120-month period of operation from start of facility commerical operation and each subsequent 120-month period of operation during which the examination or test is determined to be impractical.

(6)(i) The Commission will evaluate determinations under paragraph (g)(5) of this section that code requirements are impractical. The Commission may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

(ii) The Commission may require the licensee to follow an augmented inservice inspection program for systems and components for which the Commission deems that added assurance of structural reliability is necessary.

EVALUATION OF COMPLIANCE

Exemptions from the requirements to Section XI of the ASME Code were requested in the original submittals (References 1 and 2) and in subsequent requests to the NRC (References 6, 7, 8, 9, and 11).

Interim relief from the requirement for which exemptions were requested in the submitted program was granted by the NRC on December 7, 1977. Final NRC approval, including requested relief, of the Zion ISI program was granted on August 8, 1980.

The NRC has not yet approved proposed amendments to the Zion Technical Specifications to conform with the submitted ISI program. The Zion inservice inspections and tests will be performed either to the requirements of the submitted program or the Technical Specification whichever is most conservative, until final NRC approval of the program is received.

STATEMENT OF SECTION 50.55a - PARAGRAPH (h)

Protection systems: For construction permits issued after January 1, 1971, protection systems shall meet the requirements set forth in editions or revisions of the Institute of Electrical and Electronics Engineers Standard: "Criteria for Protection Systems for Nuclear Power Generation Stations," (IEEE-279) in effect on the formal docket date of the application for a construction permit. Protection systems may meet the requirements set forth in subsequent editions or revisions of IEEE-279 which become effective.

EVALUATION OF COMPLIANCE

The Zion construction permit was issued on December 26, 1968.

All features of the protection systems which actuate reactor trip and engineered safety features action are designed and/or built by Westinghouse and conform to the intent of the criteria specified in IEEE-279 of August 30, 1968 (FSAR, Question 7.7).

STATEMENT OF SECTION 50.55a - PARAGRAPH (i)

Fracture toughness requirements: Pressure-retaining components of the reactor coolant pressure boundary shall meet the requirements set forth in Appendices G and H to this part.

EVALUATION OF COMPLIANCE

Zion has addressed and complies with the requirements of 10 CFR Part 50, Appendices G and H as discussed in the evaluation of compliance for each of these appendices.

STATEMENT OF SECTION 50.55a - PARAGRAPH (j)

Power reactors for which a notice of hearing on an application for a provisional construction permit or a construction permit has been published on or before December 31, 1970, may meet the requirements of paragraphs (c)(1), (d)(1), (e)(1), and (f)(1) of this section instead of paragraphs (c)(2), (d)(2), (c)(2), and (f)(2) of this section, respectively.

EVALUATION OF COMPLIANCE

The Zion construction permit was issued on December 26, 1968.

Zion complies with 10 CFR 50.55a paragraphs (c)(1), (d)(1), (e)(1), and (f)(1) as discussed previously.

CONCLUSION

Zion has addressed and complies with all applicable asspects of 10 CFR 50.55a. The Zion ISI program was submitted for NRC approval on May 27, 1977. Final approval was granted on August 8, 1980. During the review period, Zion inservice inspections and tests were performed either to the submitted program or to the Technical Specifications, whichever was most conservative.

REFERENCES

FSAR Sections

Section 1.2.1	Structures - Definition of Seismic Classes
Table 4.1-11	Reactor Coolant System Boundary - Code Requirements
Chapter 5	Containment Structures
Table 6.2-1	Emergency Core Cooling System - Code Requirements
Table 9.2-1	Chemical and Volume Control System Code Requirements

Table 9.3-1	Component Cooling System - Code Requirements
Table 9.4-1	Residual Heat Removal System Code Requirements
Table 9.5-1	Spent Fuel Pit Cooling System Code Requirements
Table 9.8-1	Sampling System Code Requirements

FSAR Questions

1.5	10 CFR 50, Appendix B
4.10	Inservice Inspection
4.11	Stress Intensity Limits
4.12	Plastic Instability Analysis
4.13	Design and Fatigue Analysis Transients
4.14	Component Codes Used
4.18	Prototype Vibration Data
4.21	Emergency and Faulted Conditions Categories
4.22	Vibration Operational Test Program
4.23	Seismic Analysis of Class I Systems
4.29	Seismic Effects of Class II Piping Systems on Class I Piping
4.35	Loop Isolation Valves
4.43	Overpressure Protection
4.44	Faulted Conditions
4.45	Plastic Hinges for Piping Analysis
4.48	Emergency Operating Conditions Stress Limits
4.50	Class I Piping Stress Limits
4.64	OBE - DBE
4.65	Loading Cycles
4.66	ANSI B31.10 - ANSI B31.7.0

5.8	Thermal Effect in Concrete
5.18	ACI-318-63
5.21	Checking design
5.23	Elastic Low Strain Behavior
5.24	Base Slab
5.33	Friction Losses of Tendons
5.42	Penetration to Liner
5.44	Liner Stress
5.47	Tendon Anchorage Assemblies
5.49	ASTM-A-615 Grade 60 Welds
5.52	Tendon Ducts
5.56	Tendon Surveillance
5.60	Load Factors
5.84	Reactor Vessel Cavity Design Criteria
6.5	Paint for Containment Inner Surface
6.10	Inservice Inspection Program
6.14	Fan Cooler Motor
7.1	Interlock System
7.2	Protection and Control Systems
7.7	Protection Systems (which actuate reactor trip and engineered safety feature action)
7.11	Reactor Protection System and Engineered Safety Features Actuation System
7.14	Indication of Protections System Testing to Operator
7.20	Isolation Valve Seal Water System
7.21	Penetration Pressurization System
7.24	Auxiliary Feedwater System Controls

7.25 Motor Operated Isolation Valves

7.26 Control Circuit for Motor-Operated Isolation Valves.

Zion Station Radiological Safety Technical Specifications

Sections 3/4.3

Personal Communications

1. R. L. Bolger, CECo, letter to E. G. Case, NRC, May 27, 1977.

2. R. L. Bolger, CECo, letter to E. G. Case, NRC, June 29, 1977.

3. A. Schwencer, NRC, letter to R. L. Bolger, CECo, December 7, 1977 Re. Zion Unit 1.

4. A. Schwencer, NRC, letter to R. L. Bolger, CECo, December 7, 1977 Re. Zion Station, Unit 2.

5. D. E. O'Brien, CECo, letter to E. G. Case, NRC, December 13, 1977.

6. W. F. Naughton, CECo, letter to A. Schwencer, NRC, March 15, 1978.

7. W. F. Naughton, CECo, letter to H. R. Denton, NRC, June 28, 1979.

8. D. L. Peoples, CECo, letter to H. R. Denton, NRC, October 1, 1979.

9. A. Schwencer, NRC, letter to D. L. Peoples, CECo, October 26, 1979.

10. D. L. Peoples, CECo, letter to H. R. Denton, NRC, March 4, 1980.

10 CFR 50.59 - CHANGES, TESTS AND EXPERIMENTS

STATEMENT OF SECTION 50.59 - PARAGRAPHS (a) AND (b)

(a)(1) The holder of a license authorizing operation of a production or utilization facility may (i) make changes in the facility as described in the safety analysis report, (ii) make changes in the procedures as described in the safety analysis report, and (iii) conduct tests or experiments not described in the safety analysis report, without prior Commission approval, unless the proposed change, test or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question. (2) A proposed change, test or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

(b) The licensee shall maintain records of changes in the facility and of changes in the procedures made pursuant to this section, to the extent that such changes constitute changes in the facility as described in the safety analysis report or constitute changes in procedures as described in the safety analysis report. The licensee shall also maintain records of tests and experiments carried out pursuant to paragraph (a) of this section. These records shall include a written safety evaluation which provides the bases for the determination that the change, test, or experiment does not involve an unreviewed safety question. The licensee shall furnish to the appropriate NRC Regional Office shown in Appendix D of Part 20 of this chapter with a copy to the Director of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, annually or at such shorter intervals as may be specified in the license, a report containing a brief description of such changes, tests, and experiments, including a summary of the safety evaluation of each. Any report submitted by a licensee pursuant to this paragraph will be made part of the public record of the licensing proceeding. In addition to a signed original, 39 copies of each report of changes in the facility of the type described in Section 50.21(b) or Section 50.22 or a testing facility, and 12 copies of each report of changes in any other facility, shall be filed. The records of changes in the facility shall be maintained until the date of termination of the license, and records of changes in procedures and records of tests and experiments shall be maintained for a period of five years.

EVALUATION OF COMPLIANCE

1. Zion Station Radiological Safety Technical Specifications, page 301

The offsite review and investigative function shall review:

a. The safety evaluation for tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

2. Zion Station Radiological Safety Technical Specifications, page 314

Records of changes made to the equipment or review of tests and experiments to comply with 10 CFR 50.59 shall be kept in a manner convenient for review and shall be retained for at least five years.

3. Zion Station Radiological Safety Technical Specifications, page 317

The reporting requirements of Title 10, Code of Federal Regulations will be complied with.

CONCLUSIONS

Zion Station complies with the requirements of 10 CFR 50.59.

REFERENCES

Zion Station Radiological Safety Technical Specifications, Section 6.0, Administrative Controls, pages 300-333.

10 CFR 50.70 - INSPECTIONS

STATEMENT OF SECTION 50.70

(a) Each licensee and each holder of a construction permit shall permit inspection, by duly authorized representatives of the Commission, of his records, premises, activities, and of licensed materials in possession or use, related to the license or construction permit as may be necessary to effectuate the purposes of the Act, including Section 105 of the Act.

(b)(1) Each licensee and each holder of a construction permit shall upon request by the Director, Office of Inspection and Enforcement, provide rent-free office space for the exclusive use of the Commission inspection personnel. Heat, air conditioning, light, electrical outlets and janitorial services shall be furnished by each licensee and each holder of a construction permit. The office shall be convenient to and have full access to the facility and shall provide the inspector both visual and acoustic privacy.

(2) For a site with single power reactor or fuel facility licensed pursuant to Part 50, the space provided shall be adequate to accommodate a full-time inspector, a part-time secretary and transient NRC personnel and will be generally commensurate with other office facilities at the site. A space of 250 square feet either within the site's office complex or in an office trailer or other on site space is suggested as a guide. For sites containing multiple power reactor units of fuel facilities, additional space may be requested to accommodate additional full-time inspector(s). The office space that is provided shall be subject to the approval of the Director, Office of Inspection and Enforcement. All furniture, supplies and communication equipment will be furnished by the Commission.

(3) The licensee or construction permit holder shall afford any NRC resident inspector assigned to that site, or other NRC inspectors identified by the Regional Director as likely to inspect the facility, immediate unfettered access, equivalent to access provided regular plant employees, following proper identification and compliance with applicable access control measures for security, radiological protection and personal safety.

EVALUATION OF COMPLIANCE

Plant operating records shall be kept in a manner convenient for review (Zion Radiological Safety Technical Specifications P. 314).

Rental free office space as detailed above is provided at the Zion Station for the Resident Inspectors Office.

Plant access equivalent to that provided regular plant employees is provided to NRC inspectors identified by the Regional Director of Region III as likely to inspect the facility.

CONCLUSION

Zion Station complies with the requirements of 10 CFR 50.70.

REFERENCES

Zion Station Radiological Safety Technical Specifications Section 6.0, Administrative Controls

10 CFR 50.71 - MAINTENANCE OF RECORDS, MAKING OF REPORTS

STATEMENT OF SECTION 50.71 - PARAGRAPHS (a) THROUGH (d)

(a) Each licensee and each holder of a construction permit shall maintain such records and make such reports, in connection with the licensed activity, as may be required by the conditions of the license or permit or by the rules, regulations, and orders of the Commission in effectuating the purposes of the Act, including section 105 of the Act.

(b) With respect to any production or utilization facility of a type described in Section 50.21(b) or Section 50.22, or a testing facility, each licensee and each holder of a construction permit shall, upon each issuance of its annual financial report, including the certified financial statements, file a copy thereof with the Director of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555.

(c) Records which are required by the regulations in this part, by license condition, or by technical specification, shall be maintained for the period specified by the appropriate regulation, license condition, or technical specification. If a retention period is not otherwise specified, such records shall be maintained until the Commission authorizes their disposition.

(d)(1) Records which must be maintained pursuant to this part may be the original or a reproduced copy or microform if such reproduced copy or microform is duly authenticated by authorized personnel and the microform is capable of reproducing a clear and legible copy after storage for the period specified by Commission regulations.

(2) If there is a conflict between the Commission's regulations in this part, license condition, or technical specification, or other written Commission approval or authorization pertaining to the retention period for the same type of record, the retention period specified in the regulations in this part for such records shall apply unless the Commission, pursuant to Section 50.12, has granted a specific exemption from the record retention requirements specified in the regulations in this part.

EVALUATION OF COMPLIANCE

The maintenance of records, and making of reports is discussed in Section 6.0 of the Zion Station Radiological Safety Technical Specifications. This document is updated periodically and is subject to NRC approval.

STATEMENT OF SECTION 50.71 - PARAGRAPH (e)

Each person licensed to operate a nuclear power reactor pursuant to the provisions of 50.22 shall update periodically, as provided in paragraphs (e)(3) and (e)(4) of this section, the final safety analysis report (FSAR) originally submitted as part of the application for the operating license, to assure that the information included in the FSAR contains the latest material developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirement since submission of the original FSAR or, as appropriate, the last updated FSAR. The updated FSAR shall be revised to include the effects of: all changes made in the facility or procedures as described in the FSAR; all safety evaluations performed by the licensee either in support of requested license amendments or in support of conclusions that changes did not involve an unreviewed safety question; and all analyses of new safety issues performed by or on behalf of the licensee at Commission request. The updated information shall be appropriately located within the FSAR.

(1) Revisions containing updated information shall be submitted on a replacement-page basis and shall be accompanied by a list which identifies the current pages of the FSAR following page replacement. One signed original and 12 additional copies of the required information shall be filed with the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulation Commission, Washington, D. C. 20555.

(2) The submittal shall include (i) a certification by a duly authorized officer of the licensee that either the information accurately presents changes made since the prevous submittal, necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirement, or that no such changes were made; and (ii) an identification of changes made under the provisions of Section 50.59 but not previously submitted to the Commision.

(3) (i) A revision of the original FSAR containing those original pages that are still applicable plus new replacement pages shall be filed within 24 months of either July 22, 1980, or the date of issuance of the operating license, whichever is later, and shall bring the FSAR up to date as of a maximum of 6 months prior to the date of filing the revision.

(ii) No less than 15 days before Section 50.71(e) becomes effective, the Director of the Office of Nuclear Reactor Regulation shall notify by letter the licensees of those nuclear power plants initially subject to the NRC's systematic evaluation program that they need not comply with the provisions of this section while the program is being conducted at their plant. The Director of the Office of Nuclear Reactor Regulation will notify by letter the licensee of each nuclear power plant being evaluated when the systematic evaluation program has been completed.

Within 24 months after receipt of this notification, the licensee shall file a complete FSAR which is up to date as of a maximum of 6 months prior to the date of filing the revision.

(4) Subsequent revisions shall be filed no less frequently than annually and shall reflect all changes up to a maximum of 6 months prior to the date of filing.

(5) Each replacement page shall include both a change indicator for the area changed, e.g., a bold line vertically drawn in the margin adjacent to the portion actually changed, and a page change identification (date of change or change number or both).

EVALUATION OF COMPLIANCE

Commonwealth Edison Company shall submit a revision of the original Zion Station FSAR within 24 months of July 22, 1980 and shall submit subsequent revisions to this document as required by this paragraph.

CONCLUSIONS

Zion Station complies with the requirements of 10 CFR 50.71 with the exception that an updated FSAR has not yet been submitted. This document shall be submitted in accordance with the schedule established by the NRC in Section 50.71 - Paragraph (e).

REFERENCES

Zion Station Radiological Safety Technical Specifications, Section 6.0, Administrative Controls, pages 300-333.

APPENDIX A TO 10 CFR 50 - GENERAL DESIGN CRITERIA (GDC)

OVERALL REQUIREMENTS

STATEMENT OF GDC 1 - QUALITY STANDARDS AND RECORDS

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 1. Systems, structures, and components have been designed, fabricated, erected, and tested to quality levels commensurate with their relationship to safety. The appropriate codes employed for various items have been supplemented where required. A quality assurance program has been employed and appropriate records have been and are being maintained directly by the Commonwealth Edison Company (Commonwealth Edison) or are under Commonwealth Edison control.

2. General Review

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

Design, Fabrication, Erection, and Testing

a. FSAR, page 1.2-2

1. Seismic Class I - Failure would cause or increase the severity of a LOCA; vital to safe shutdown and isolation.

2. Seismic Class II - Function in direct support of reactor operation.

3. Seismic Class III - Neither Class I or II.

b. SER, Section 3.7, page 3-10

1. Containment structures - We conclude that the provisions for testing and surveillance of the containment are acceptable.

2. Class I structures other than containment - The information provided by the Applicant with respect to the design of these structures has been evaluated and found to be consistent with that provided for previously approved (similar) facilities and therefore acceptable for the Zion Station.

c. SER, Section 3.9, page 3-11

All electrical systems and components vital to plant safety, including the diesel generators, are designed to Class I standards so their integrity would not be impaired by the DBE. We conclude that the seismic design of Class I instrumenation and electrical equipment is acceptable.

d. FSAR, page 1.3-1

Those features of the reactor facility which are essential either to the prevention of accidents that could affect the public health and safety or to the mitigation of their consequences are designed, fabricated, and erected to the following standards:

1. Quality standards that reflect the importance of the safety function to be performed.

2. Performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquakes, flooding conditions, winds, ice, or other natural phenomena characteristic of the Zion site.

e. FSAR, Appendix A, page A-1

Those systems and components of reactor facilities which are essential either to the prevention of accidents that could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed.

Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with required safety functions.

Identification of Codes and Standards and Their Evaluations

a. SER, Section 3.2, page 3-2

We find these classifications to be acceptable and have concluded that the Applicant placed all safety-related structures, systems, and components in their appropriate category.

b. SER, Section 3.8, page 3-10

We find the codes and standards specified for Class I tanks, heat exchangers, piping, pumps, and valves provide an acceptable quality level and are consistent with those proposed and approved for recently reviewed plants of this type.

c. FSAR, page 1.3-1

1. The concrete structure of the reactor containment conforms to the applicable portions of ACI-318-63.

2. Vessels comply with the ASME Boiler and Pressure Vessel Code under the specific classification dictated by their use or appropriate codes.

3. In the same manner, piping conforms to the requirements of the USA Standard Code for Pressure Piping (ASA B31.1-1955) and Nuclear Code Cases N-7 and N-10.

d. FSAR, Appendix A, page A-1

Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified.

e. FSAR, Appendix A, page A-1

Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety functions, they shall be supplemented or modified as necessary.

A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions.

a. SER, page 18-3

The Applicant has undertaken to meet the standards set forth in ANSI N45.2, "Quality Assurance Requirements for Nuclear Power Plants," and ANSI N18.7, "Standard for Administrative Controls for Nuclear Power Plants."

b. FSAR, page 1.9-1

A comprehensive quality assurance program has been instituted by Commonwealth Edison for the design and construction of the Zion Station.

c. FSAR, page Q1.9-23

Zion Station has been designed and constructed under a quality assurance program developed by Commonwealth Edison. This program has been in effect since 1968. The basic quality areas covered by the Commonwealth Edison program are the same as those presented in 10 CFR 50 Appendix B and in ANSI N45.2-1971. The Commonwealth Edison program has effectively assured that the requisite quality levels have been incorporated into the design and construction of Zion Station.

d. Commonwealth Edison Company Quality Assurance Program for Nuclear Generating Stations, January 1976, page 1-1

Implementation of the program with quality procedures provides the degree of quality assurance commensurate with the requirements of ASME Section III, Division 1 and Division 2 for concrete containment and other applicable codes, Nuclear Regulatory Commission requirements, and federal regulations governing design, procurement, construction, testing, operation, refueling, maintenance, repair, and modification of Commonwealth Edison's nuclear power generating facilities.

e. Topical Report Evaluation: letter to W. B. Behnke from R. H. Vollmer, dated December 29, 1975

We have reviewed and evaluated the Commonwealth Edison Quality Assurance Program for Nuclear Generating Stations (Topical Report CE-1, June 1975). We find that it describes an acceptable quality assurance program for the design, procurement, construction, and operation activities that are within the Commonwealth Edison scope of work for nuclear power plants.

f. Topical Report Evaluation as Reviewed by the NRC Quality Assurance Branch, page 2

Regulatory Position

It is the staff's position that the Commonwealth Edison Quality Assurance Program for Nuclear Generating Stations (Topical Report CE-1, June 1975) is acceptable for use in the design, procurement, construction, and operation of nuclear power plants.

Appropriate records of the design, fabrication, erection, and testing of structures, systems and components important to safety shall be maintained by or under the control of nuclear power unit throughout the life of the unit.

Records

a. FSAR page Q1.5-48

Records are retained and maintained in accordance with a Quality Procedure listed in Appendix 3 of Question 1.5 in the FSAR to furnish evidence of activities affecting quality.

b. FSAR, Appendix A, page A-1

A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

c. FSAR, page Q1.5-48

Records are retained and maintained in accordance with a quality procedure list in Appendix 3 to furnish evidence of activities affecting quality.

3. Conclusion

All aspects of GDC 1 were addressed in the Zion design.

4. References

FSAR Sections

1.2.1 Structure Seismic Criteria

1.3 General Design Criteria

1.9 Quality Assurance

Appendix B - Criteria for Vessels and Piping within Reactor Coolant System Pressure Boundary

FSAR Questions

1.1 Quality Assurance Criteria

1.2 Component Quality Control

1.5 List of Seismic Class I Structures, Systems, and Components (Appendix 1)

1.9 Safety Guide 28 - Quality Assurance Program Requirements (Design and Construction)

4.7 Reactor Coolant Loop Piping Weld Quality Assurance

4.14 .eactor Coolant System Code Requirements

4.33 Seismic Design Basis Quality Assurance Methods

5.17, 5.18 Containment Stress Code

5.21 Quality Assurance Criteria for ACI Code

5.22 Containment Structural Strength Integrity Assurance

5.25 Surveillance of Containment Structure

7.4 Test Documentation of Safety Related Electrical Equipment

7.9 Reactor Protection System Quality Assurance

8.16 Cable Installation Surveillance

10.2 MSIV Acceptance Criteria

11.9 Radiation Survey's Acceptance Standards

11.38 Quality Control of Radwaste Counting Equipment

SER Sections

3.1 Conformance with AEC General Design Criteria

3.2 Classification of Structures, Components, and Systems

3.6.3 Seismic Instrumentation

3.6.4 Seismic Design Control Measures

3.7 Design of Class I Structures

3.7.1 Containment Structure

3.7.2 Class I Structures Other than Containment

3.8 Mechanical Systems and Components

3.9 Seismic Design of Class I Instrumentation and Electrical Equipment

18.0 Quality Assurance

STATEMENT OF GDC 2 - DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 2. The systems, components, and structures important to safety have been designed to accommodate, without loss of capability, the most severe natural phenomena recorded and appropriate combinations of postulated accidents with natural phenomena. The importance of the safety functions of the various items has been considered.

2. General Review

Safety-related structures, systems, and components shall be designed to withstand the effects of natural phenomena.

Earthquakes

a. Design (FSAR page 2.11-2)

   1. OBE: 0.08g horizontal acceleration
      0.05g vertical acceleration

   2. DBE: 0.17g horizontal acceleration
      0.11g vertical acceleration

b. Seismic classification of structures, systems, and components (FSAR, page 1.2-2)

   1. Seismic Class I - Failure would cause or increase the severity of a LOCA; vital to safe shutdown and isolation.

   2. Seismic Class II - Function in direct support of reactor operation.

3. Seismic Class III - Neither Class I or II.

c. List of Seismic Class I systems, structures, and components (FSAR Question 1.5, Appendix 1)

d. NRC evaluation of seismic design (SER Appendix E, October 6, 1972)

"On the basis of the information presented by the applicant, it is our opinion that the approach to the seismic analysis and design of the Zion Station Units 1&2 has resulted in a design that is adequate to resist the earthquake conditions postulated at the site."

Tornadoes

a. Containment tornado loading (FSAR, page 5.1-12)

1. three-psid design pressure;

2. tangential velocity of 300 mph, forward progression of 60 mph; and

3. tornado driven missile - 8-in. diameter 12-foot long piece of wood at 225 mph.

b. All Seismic Class I structures have been analyzed to show capability to withstand tornado missiles (Question 5.12)

Hurricanes

Hurricanes are not specifically addressed. However, Zion is designed for a 95 mph wind load (SER, page 3-2).

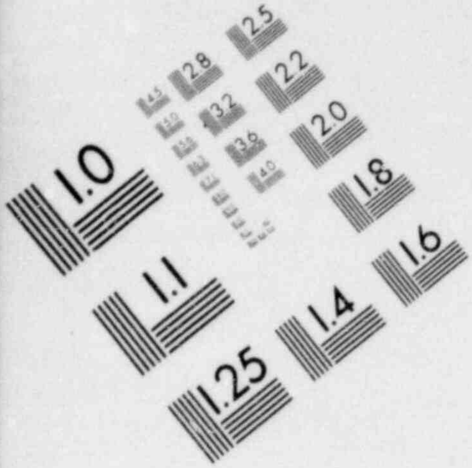
Floods

The maximum water level, including wave activity, was estimated to be below plant grade. The applicant has concluded, therefore, that such an event would not adversely affect any safety-related systems (SER, page 2-13).

Tsunami & Seiche

We conclude that all bases for site flooding from tsunami and seiche are acceptable and that adequate provisions have been made to assure no loss of safety-related functions (SER, page 2-14).

Combinations of effects of normal and accident conditions with the effects of natural phenomena shall be considered in the design of safety-related structures, systems, and components.



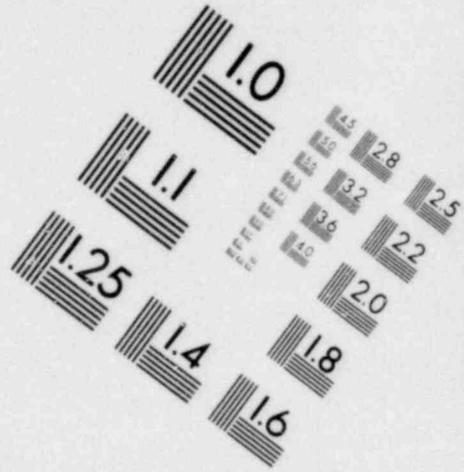
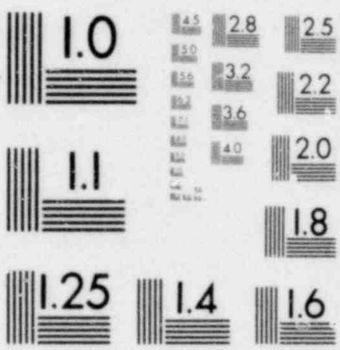
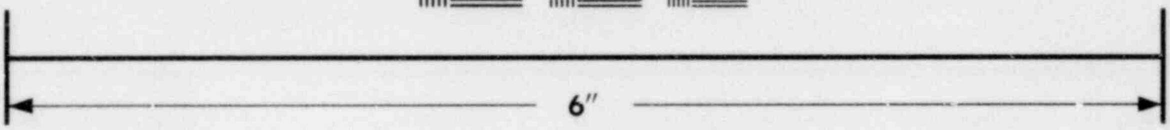
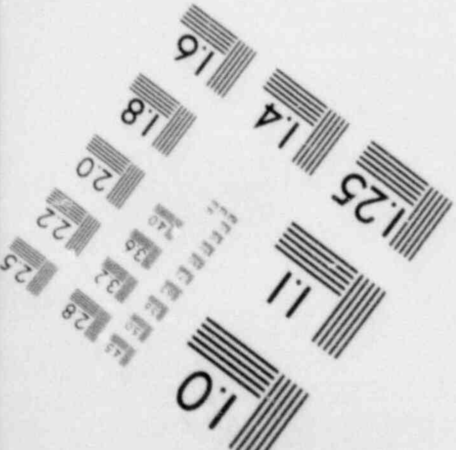


IMAGE EVALUATION
TEST TARGET (MT-3)







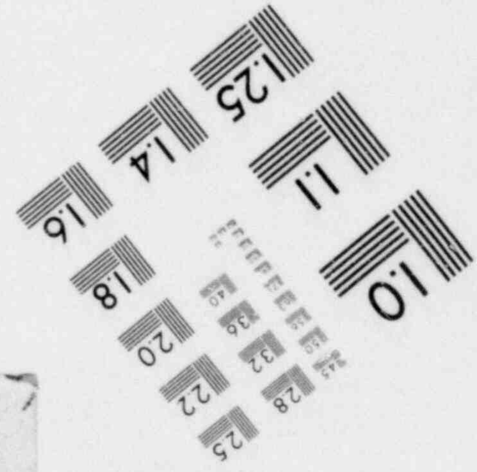


IMAGE EVALUATION
TEST TARGET (MT-3)

6″

Load combinations are discussed in FSAR Section 5.1.2.4, Question 5.68, Table 5.1-2, and Question 5.4.

3. Conclusion

All aspects of GDC 2 were addressed in the Zion design.

4. References

FSAR Sections

1.2.1	Structures Seismic Criteria
2.6.2.1	Currents, Tides, Waves and Littoral Drift
2.6.2.3	Floods
2.7.3.5	Severe Weather
2.10	Seismology
2.11.1	Plant Design Basis Regarding Site and Environment
4.14	Design Characteristics - Seismic Loads
4.2.4	Protection Against Proliferation of Dynamic Effects - Tornado Protection
4.3.1	Safety Factors - Piping - Seismic Loads - Combined Blowdown and Seismic Loads
5.1.2.2	Design Load Criteria (tornado load criteria included)
5.1.2.4	Structural Design Basis - At Design Loads
Table 5.1-1	Load Case Summary
Table 5.1.2	Summary of Concrete and Reinforcing Steel Stress
7.2.1	Protection System Seismic Performance

FSAR Questions

1.5	List of Seismic Class 1 Structures, Systems, and Components (Appendix 1)
2.1, 2.30	Probable Maximum Seiche
2.22, 2.29	Probable Maximum Flood on Bull Creek

2.24	DBE Liquefaction of Soil
2.26	Earthquake Generated Seiche
2.32	Ice
4.23, 4.59	Seismic Analysis of Seismic Category I Components
4.5, 4.24	Combination of Seismic Loads at a Node - Root Mean Square and Absolute Sum
4.25, 4.27, 4.54, 4.69	Floor Response Spectra
4.26, 4.52, 4.67	Seismic Design Basis, Vertical Load Factors
4.28	Check of Seismic System Analysis - Time-History Analysis vs. Response Spectrum Analysis
4.29, 4.60	Seismic Effects of Class II Piping on Class I Piping
4.30	Seismic Design Criteria for Seismic Class I Piping Outside Containment
4.31, 4.55	Seismic Design Criteria for Class I Components
4.32	Field Location of Seismic Supports
4.33	Seismic Design Review
4.44	Faulted Conditions - Load Combinations
4.48, 4.64	Emergency Operation Condition Stress Limits
4.49, 4.65	Earthquake Cycles
4.50	Stress Limits
4.53, 4.68	Analysis of Seismic Class I piping
4.57	Equipment Seismic Design Criteria
4.58	Damping Ratios
5.2	Design Underground Facilities for Earthquake
5.3, 5.79, 5.83	Dynamic System Analysis
5.4, 5.5	Seismic Category I Structures

5.6, 5.76	Torsional Modes in Seismic Analysis
5.7, 5.77	Soil Structure Interactions
5.10, 5.58	Class II (Seismic) Structures - Effects on Class I Structures
5.11	Tornado Protection
5.12	Missiles - Tornado Design Basis Missiles
5.15	Missiles - Tornado Generated
5.16	Equipment Tie Down Criteria - Resistance to Seismic and Tornado Forces
5.20	Stress Analysis - Load Combination
5.24	Stress Analysis - Base slab
5.25, 5.32	Stress Analysis - Computer Codes
5.34	Base Slab - Stress Calculations
5.35	Earthquake Shears
5.38	Seismic Shear - Friction
5.39	Dimension of Reinforcement
5.60	Load Factors
5.61	Tornado Generated Missiles
5.62	Seismic Design of Symmetrical Building
5.68	Load Combination
5.73	LOBAR Motion
5.80	Foundations - Load Combination
5.85	Seismic Cable Tray Design
5.86	Seismic Deflections
7.5	Seismic Evaluation Temperature Detectors
7.9	Seismic Design Criteria for RPS
8.12	Seismic Testing of DC Systems
8.26	Seismic Design Criteria for Cable Trays

10.6	Ice Blockage of Intake
11.10	Seismic Classification of Radwaste Systems

SER Sections

2.4.2	Floods
2.5	Geology and Seismology
3.2	Classification of Structures, Components and Systems
3.3	Wind and Tornado Design Criteria
3.4	Water Level (Flood) Design Criteria
3.6	Seismic Design
3.7	Design of Class I Structures
3.8	Mechanical Systems & Mechanical Components
4.2.2.2	Dynamic System
5.2.2	System Quality Group Classifications
Appendix D	Report of the Coastal Engineering Research Center - Department of the Army (Evaluation of Design Water Levels)
Appendix E	Report of John A. Blume & Associates, Engineers (Seismic Design Evaluation)

STATEMENT OF GDC 3 - FIRE PROTECTION

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertant operation does not significantly impair the safety capability of these structures, systems, and components.

EVALUATION OF COMPLIANCE

1. General Review

A detailed, comprehensive review of fire protection at Zion Station has been performed and is recorded in the following documents:

a. Fire Protection Report, April 29, 1977;

b. The fire protection and review team's site visit of July 11-14, 1977 and September 28-29, 1977; and

c. The Licensee's response to requests for additional information and staff positions.

2. Conclusion

All aspects of GDC 3 have been addressed in the original Zion design and in the subsequent fire protection review.

3. References

a. Fire Protection Report in Response to Appendix A of BTP APCSB, April 29, 1977, page 4.5-1.

b. The Fire Protection Review Team's Site Visit of July 11-14, 1977, and September 28-29, 1977.

c. Safety Evaluation Report on Fire Protection, March 1978.

d. D. E. O'Brien, Commonwealth Edison, letter to A. Schwencer, NRC, January 13, 1978.

e. D. L. Peoples, Commonwealth Edison, letter to H. R. Denton, NRC, May 28, 1980.

f. D. L. Peoples, Commonwealth Edison, letter to H. R. Denton, NRC, April 30, 1980.

g. W. F. Naughton, Commonwealth Edison, letter to H. R. Denton, NRC, April 14, 1980.

h. W. F. Naughton, Commonwealth Edison, letter to E. Reeves, NRC, March 10, 1980.

i. W. F. Naughton, Commonwealth Edison, letter to H. R. Denton, NRC, October 31, 1979.

j. W. F. Naughton, Commonwealth Edison, letter to H. R. Denton, NRC, August 31, 1979.

k. W. F. Naughton, Commonwealth Edison, letter to H. R. Denton, NRC, July 27, 1979.

l. C. Reed, Commonwealth Edison, letter to H. R. Denton, NRC, July 27, 1979.

m. C. Reed, Commonwealth Edison, letter to H. R. Denton, NRC, June 6, 1979.

n. W. F. Naughton, Commonwealth Edison, letter to H. R. Denton, NRC, May 23, 1979.

o. W. F. Naughton, Commonwealth Edison, letter to H. R. Denton, NRC, April 12, 1979.

p. W. F. Naughton, Commonwealth Edison, letter to H. R. Denton, NRC, April 9, 1979.

q. C. Reed, Commonwealth Edison, letter to H. R. Denton, NRC, March 14, 1979.

r. A. Schwencer, NRC, letter to C. Reed, Commonwealth Edison, February 14, 1979.

s. W. F. Naughton, Commonwealth Edison, letter to H. R. Denton, NRC, January 31, 1979.

t. W. F. Naughton, Commonwealth Edison, letter to Director of Nuclear Reactor Regulation, September 29, 1978.

u. W. F. Naughton, Commonwealth Edison, letter to Director of Nuclear Reactor Regulation, September 8, 1978.

v. W. F. Naughton, Commonwealth Edison, letter to Director of Nuclear Reactor Regulation, July 27, 1978.

w. W. F. Naughton, Commonwealth Edison, letter to Director of Nuclear Reactor Regulation, July 11, 1978.

x. M. S. Turbak, Commonwealth Edison, letter to G. E. Lear, NRC, June 29, 1978.

y. A. Schwencer, NRC, letter to C. Reed, Com onwealth Edison, June 26, 1978.

z. V. Stello, NRC, letter to C. Reed, Commonwealth Edisor, June 5, 1978.

aa. A. Schwencer, NRC, letter to C. Reed, Commonwealth Edison, June 1, 1978.

bb. G. Lear, NRC, letter to C. Reed, Commonwealth Edison, May 26, 1978.

cc. W. F. Naughton, Commonwealth Edison, letter to Director of Nuclear Reactor Regulation, May 1, 1978.

dd. W. F. Naughton, Commonwealth Edison, letter to E. G. Case, NRC, April 14, 1978.

ee. W. F. Naughton, Commonwealth Edison, letter to A. Schwencer, NRC, March 15, 1978.

ff. C. Reed, Commonwealth Edison, letter to A. Schwencer, NRC, January 30, 1978.

gg. D. E. O'Brien, CECO, letter to A. Schwencer, NRC, January 24, 1978.

hh. A. Schwencer, NRC, letter to R. L. Bolger, Commonwealth Edison, December 21, 1977.

ii. D. E. O'Brien, Commonwealth Edison, letter to A. Schwencer, NRC, December 14, 1977.

jj. D. E. O'Brien, Commonwealth Edison, letter to A. Schwencer, NRC, November 18, 1977.

kk. A. Schwencer, NRC, letter to R. L. Bolger, Commonwealth Edison, August 19, 1977.

ll. R. L. Bolger, Commonwealth Edison, letter to E. G. Case, NRC, July 21, 1977.

mm. D. K. Davis, NRC, letter to R. L. Bolger, Commonwealth Edison, June 24, 1977.

STATEMENT OF GDC 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1977

The Zion Station design conforms with the intent of Criterion 4. The safety-related systems, components, and structures are designed to accommodate all normal or routine environmental conditions as well as those associated with postulated accidents (where appropriate). The design includes provisions to protect, where appropriate, those safety-related items from dynamic effects resulting from component failures and specific credible outside events and conditions.

2. General Review

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

a. Environmental design of engineered safety feature equipment (SER, page 3-12)

   We find that the features provided in the design for protection against environment effects are acceptable.

b. Site and environmental structures: seismic criteria (FSAR, Section 1.2.1, page 1.2-2)

   For Seismic Class I equipment, dynamic methods were used to determine that components and structures will operate or maintain their integrity, as required.

c. Engineered safety features components capability (FSAR, page 6.1-7)

   Instrumentation, motors, cables, and penetrations located inside the containment are selected to meet the most adverse accident conditions to which they may be subjected.

d. (FSAR, page 6.1-8)

Protection, in the form of barriers, restraints, supports, and physical separation has been provided to assure that in the unlikely event of an accident the following criteria will be met:

1. Cor ainment integrity will be maintained thruughout the accident.

2. A second accident will not occur as a resul* of the original accident.

3. Sufficient safety features will be available to control the accident and safely shut the plant down.

These structures, systems and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

a. (FSAR, Sectic.1 4.2.4, page 4.2-22)

Protection has been provided against the following dynamic effects:

1. Jet forces resulting from the release of high pressure steam or water from a ruptured line,

2. Pipe whip caused by the formation of a plastic hinge in a pipe due to a rupture somewhere else in the same pipe, and

3. Missiles which can be generated in coincidence with an accident.

b. (FSAR, page 4.2-24)

All essential equipment has either been designed to withstand a credible tornado, including a single large missile generated thereby, or has been placed in a structure that will withstand the tornado and missile.

c. (FSAR, page 5.1-44)

Missile protect.on for the contain..ent liner is provided to comply with the following criteria:

1. The containment and liner are protected from loss of function due to damage by such missiles as might be generated in a loss-of-coolant accident

for break sizes up to and including the double-ended severance of a reactor coolant pipe.

2. Components required to maintain containment integrity to meet the site criteria of 10 CFR 100 are protected against loss of function due to damage by missiles.

d. (FSAR, Section 6.1.1, page 6.1-4)

A loss-of-coolant accident or other plant equipment failure might result in dynamic effects or missiles. For engineered safety features which are required to ensure safety in the event of such an accident or equipment failure, protection is provided primarily by the provisions which are taken in the design to prevent the generation of missiles. In addition, protection is also provided by the layout of the plant equipment or by missile barriers in certain cases.

e. (FSAR, page Q5.12-1)

dequate protection was provided so that the containment liner was completely protected against missile impact.

f. (FSAR, Appendix A, page A-19)

Engineered safety features will be provided against dynamic effects and missiles resulting from equipment failures. The means for accomplishing this protection are described in Chapters 5, 6, and 14.

g. (SER, Section 3.5, page 3-2)

The effects of tornado-created missiles, missiles originating from failure of rotating machinery that could be subjected to overspeed, and missiles that could orig nate from failure of high-pressure piping were considered in the design. The design criteria required that there be no loss of function of a Class I structure as a result of missile action.

h. (FSAR, page Q2.28-1, Report 2)

The results of this study show that the Zion Station is designed to allow an orderly plant shutdown in the event of aircraft impact on critical plant areas and that the probability of smoke and fumes entering those critical plant areas is approximately 7×10^{-7}.

3. Conclusion

All aspects of GDC 4 were addressed in the Zion Station design.

4. References

FSAR Sections

1.2.1	Site and Environmental Structures: Seismic Criteria
2.11.1	Plant Design Bases Dependent Upon Site and Environment
2.28	Waukegan Memorial Airport
4.1.4	Reactor Coolant System Design Characteristics
4.2.4	Protection Against Proliferation of Dynamic Effects
4.3.3	System Integrity
5.1.2.6	Containment Missile Protection
6.1.1	Engineered Safety Features Criteria

FSAR Questions

2.20	Boat Accidents Near the Site
2.28	Potential Hazard from Aircraft
4.23, 4.24	Seismic Design Criteria for Primary System
5.1	Flooding Prevention for Class I Equipment
5.11	Tornado Protection for Class I Structures
5.16	Seismic and Tornado Design Criteria for Class I Structures Outside Containment
11.11	Primary Water Storage Tanks Tornado Design Criteria

SER Sections

3.10	Environmental Design of Engineered Safety Features Equipment

STATEMENT OF GDC 5 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structures, systems, and components important to safety shall not be shared between nuclear power units unless it is shown that their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 5. As shown in the FSAR, Chapter 1, safety-related systems, structures and components are not shared unless such sharing has no significant adverse impact on safety functions.

2. General Review

Sharing of Structures, Systems and Components

a. List of Shared Systems and Components FSAR, Table 1.3-1)

b. FSAR, Section 1.3.1.3, page 1.3-3)

The criteria followed in designing the two-unit station is that each unit shall operate independently of the other and a malfunction of equipment or operator error in one unit will not initiate a malfunction or error in the other unit nor affect the continued operation of the other unit. Certain auxiliary and support systems share equipment and these are identified in Table 1.3-1.

c. (FSAR, Appendix A, page A-3)

As noted in Chapter 1, those systems or components which are shared, either between the two units or functionally within a single unit, are designed in such a manner that plant safety is not impaired by the sharing.

3. Conclusions

All aspects of GDC 5 were addressed in the Zion Station design.

4. References

FSAR Sections

1.2.4 Waste Disposal System

1.3.1.3 Shared Facilities and Equipment

1.3.7 Engineered Safety Features

Table 1.3-1 List of Shared Systems and Components

9.1.1 Auxiliary and Emergency Systems Criteria: Sharing of Systema

FSAR Questions

11.1 Redundancy and Independence of Effluent Discharge Monitors

PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

STATEMENT OF GDC 10 - REACTOR DESIGN

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 10. Appropriate fuel margins are included in the plant design.

2. General Review

Specified fuel design limits shall not be exceeded during normal operation or anticipated operational occurrences.

a. FSAR, page 3.1.2-1

The reactor core, with its related control and protection system, is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations, including effects of the loss of reactor coolant flow, loss of normal feedwater, loss of offsite power, startup of an inactive reactor coolant loop, loss of external electric load, etc.

b. FSAR, page 3.1.2-2

Fuel is designed so that the following conservative limits are not exceeded during normal operation of any anticipated transient condition.

1. DNB ratio ≥ 1.30,

2. Fuel center temperature below melting point of UO_2,

3. Internal gas pressure < nominal external pressure,

4. Clad stress < zircaloy yield strength,

5. Clad strain < 1%, and

6. Cumulative strain fatigue cycles < 80% of design strain fatigue life.

3. Update of FSAR Review

Since issuance and approval of the Zion FSAR, Westinghouse has revised its design limit on internal fuel element pressure. The NRC staff has reviewed and approved this change. The revised criteria modifies Item 2.b.3 above to read as follows:

> "...to a value below that which could cause (1) the fuel-clad diametral gap to increase due to outward creep during steady-state operation and (2) extensive DNB propagation to occur."

4. Conclusion

The Zion design addresses the criteria contained in GDC 10.

5. References

FSAR Sections

1.2.3 Reactor and Plant Control

1.3 General Design Criteria

1.4 Design Parameters and Plant Comparison

1.5 Design Highlights

1.6 Research and Development Requirements

3.1 Reactor Design Basis

3.2 Reactor Design

6.2 Emergency Core Cooling System

7.3.2 Reactor Control System Design

7.4 Nuclear Instrumentation System Design and Evaluation

7.5 Engineered Safety Features Instrumentation

7.6 In-Core Instrumentation

9.4 Residual Heat Removal System

9.7 Reactor Components and Fuel Handling System

14.3.3 Core and Internals Integrity Analysis

FSAR Questions

3.1 Consequences of Single Continuous Rod Withdrawal

3.2 Power Tilt Design

3.3 Peaking Factors Analysis

3.5 Operating Procedures for Out of Service Power Tilt Monitor

7.8 Reactor control Systems

7.13 Primary System Control Room Monitors

7.25 Residual Heat Removal Design Criteria

9.1 Cooling Water Adequacy for Reactor Shutdown

SER Sections

Chapter 4.0 Reactor

STATEMENT OF GDC 11 - REACTOR INHERENT PROTECTION

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 11. A negative reactivity coefficient is a basic feature of the reactor core design.

2. General Review

Prompt inherent nuclear feedback characteristics compensate for rapid increase in reactivity. The various contributions to the power coefficient are given in FSAR Table 3.2.1-1. The overall power coefficient is negative under normal operating conditions throughout core life.

3. Conclusion

The Zion design addresses the criteria contained in GDC 11.

4. References

FSAR Sections

3.1.2 Reactor Core Design

3.2.1 Reactivity Coefficients

7.2.1 Protection Systems

14.1 Core and Coolant Boundary Protection Analysis

14.2 Standby Safeguards Analysis

FSAR Questions

1.4 Design Capability of Reactor Shutdown Outside Control Room

3.4 ΔT Trip Calibration Frequency

7.7 Reactor Protection System

7.9 Seismic Design Criteria for Reactor Protection System

7.10 Independence Criteria for Redundant Reactor Protection System

7.11 Reactor Protection System Testing Capability

7.14 Reactor Protection System Control Room Status

7.28 Radiation Effects Design Criteria for Reactor Protection System

SER Sections

4.3 Nuclear Design

7.2 Reactor Protection System

STATEMENT OF GDC 12 - SUPPRESSION OF REACTOR POWER OSCILLATIONS

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscilla'ions which can result in conditions exceeding specified fuel design limits are not possible or can be reliably and readily detected and suppressed.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 12. The design includes provisions to detect and control those power oscillations which might exceed acceptable fuel design limits during operation.

2. General Review

Power oscillations are reliably and readily detected. Out-of-core instrumentation is provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon-induced oscillations (FSAR, page 3.1.2-3).

Oscillations are reliably and readily suppressed. The potential for possible spatial oscillations of power distribution for this core has been reviewed. It is concluded that low frequency xenon oscillations may occur in the axial dimension, butany such oscillations can be suppressed by controlling the axial neutron flux by adjusting the position of Control Rod Bank D and by adjusting the moderator temperature.

3. Conclusion

The Zion design addressed the criteria contained in GDC 12.

4. References

FSAR Sections

3.1.2 Suppression of Power Oscillations

3.1.3 Reactivity Control Limits

3.2.1.1 Reactivity Control Aspect of Reactor Design

7.2.1 Redundancy of Reactivity Control

FSAR Questions

3.2 Reactor Power Tilt Analyses

3.5 Out of Service Requirements for Power Tilt Monitor

SER Sections

4.3 Nuclear Design

STATEMENT OF GDC 13 - INSTRUMENTATION AND CONTROL

Instrumentation and control shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 13. Appropriate instrumentation and control systems have been provided to monitor and control pertinent variables and systems over normal and postulated accident conditions.

2. General Review

Instrumentation and controls are provided to monitor and maintain all operationally important reactor operating parameters such as neutron flux, system pressures, flow rates, temperatures, levels and control rod positions within prescribed operating ranges. The quantity and types of instrumentation provided are adequate for safe and orderly operation of all systems and processes over the full operating range of the plant (FSAR, page 7.1-1).

3. Conclusion

The Zion design addresses the criteria contained in GDC 13.

4. References

FSAR Sections

Chapter 7 Instrumentation and Control

FSAR Questions

7.1 through 7.33 Instrumentation and Control

11.16 Effluent Release Instrumentation

SER Sections

7.0 Instrumentation and Control Systems

STATEMENT OF GDC 14 - REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 14. The design, fabrication, erection, and testing employed on the Zion reactor coolant pressure boundaries and the extensive quality control measures employed during each of the above phases insures that these pressure boundaries have extremely low probabilities of abnormal leakage, rapidly propagating failure, and gross rupture.

2. General Review

The RCP boundary has an extremely low probability of abnormal leakage, failure, or rupture.

a. FSAR, page 4.1-5

The reactor coolant system in conjunction with its control and protective provisions is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated systems interactions, and maintain the stresses within applicable code stress limits.

b. FSAR, page 4.1-6

Positive indications are provided in the control room to alert the operator of leakage of coolant from the reactor coolant system.

c. FSAR, page 4.1-8

The reactor coolant pressure boundary is designed to reduce to an acceptable level the probability of a rapidly propagating failure.

d. FSAR, page 4.1-7

The reactor coolant boundary is shown to be capable of accommodating without overpressure, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection which is considered the worst credible case.

e. FSAR, page 4.1-23

All primary pressure-containing components of the reactor coolant system are designed, fabricated, inspected, and tested in conformance with the applicable codes listed in FSAR Table 4.1-11, and are Class I seismic design.

3. Conclusion

The Zion design addresses the criteria contained in GDC 14.

4. References

FSAR Sections

1.3.6 Reactor Coolant Pressure Boundary

4.1.3 Reactor Coolant System Pressure Boundary Design Criteria

4.3.4 Reactor Coolant System Pressure Relief

14.1 Core and Coolant Boundary Protection Analysis

FSAR Questions

4.3 Reactor Coolant System Pressure Boundary Material Data

4.5 Heat Treatment

4.6 Material Metallurgy

4.7 Welding Methods

4.10 Inservice Inspection

4.11 Piping Stress Design Criteria

4.13 Design and Fatigue Analysis

4.15 Pipe Rupture Design Criteria

4.16 Piping Protection Design

4.17 RCS Component Supports Design Criteria

4.20 Pressure-relieving Design Criteria

4.22 Piping Vibrations

4.23 Class I Seismic Components of Reactor Coolant System

SER Sections

3.8 Mechanical Systems and Mechanical Components

5.2 Integrity of Reactor Coolant Pressure Boundary

STATEMENT OF GDC 15 - REACTOR COOLANT SYSTEM DESIGN

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 15. The design of the reactor coolant and associated pertinent systems includes sufficient margin to insure that the appropriate design limits of the reactor coolant pressure boundary are not exceeded during normal operation, including transients as defined in Section 14.1 of the FSAR.

2. General Review

The RCS and associated systems are designed to assure that design conditions are not exceeded.

a. FSAR, page 4.1-5

The reactor coolant system is protected from overpressure by means of pressure-relieving devices, as required by Section III of the ASME Boiler and Pressure Vessel Code. Sections of the system which can be isolated are provided with overpressure relieving devices such that the system code allowable relief pressure within the protected system is not exceeded.

b. FSAR, page 4.1-11

The design temperature for each component is selected to be above the maximum coolant temperature in that component under all normal and anticipated transient load conditions.

c. FSAR, page 4.1-12

The criteria adopted for seismic analyses for equipment are defined in Section 1.2.1. Design and construction practices in accordance with these criteria assure the integrity of the reactor coolant system under seismic loading.

3. Conclusion

The Zion design addresses the criteria contained in GDC 15.

4. References

FSAR Sections

1.3.6 Reactor Coolant Pressure Boundary

Chapter 4.0 Reactor Coolant System

14.1 Core and Coolant Boundary Protection Analysis

Appendix B Criteria for Vessels and Piping Within Reactor Coolant System Pressure Boundary

FSAR Questions

4.1 through 4.71 Reactor Coolant System

SER Sections

3.8 Mechanical Systems and Mechanical Components

5.0 Reactor Coolant system

STATEMENT OF GDC 16 - CONTAINMENT DESIGN

Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 16. The Zion containments have been furnished with pressurized channels over all liner welds, with pressurized penetrations, and with isolation valve seal water systems. These provisions insure that the containments are "essentially leak tight." Redundant and diverse means are provided (in the fan cooler and containment spray systems) to insure that containment design conditions are not exceeded in the event of a postulated accident.

2. General Review

Containment is provided to establish an essentially leaktight barrier.

a. FSAR, page 5.1-4

The reactor containment completely encloses the entire reactor and reactor coolant system and assures that essentially no leakage of radioactive materials to the environment would result even if gross failure of the reactor coolant system were to occur.

b. FSAR, page 5.2-12

The containment design leak rate is not more than 0.1 percent of the contained volume in 24 hours at 47 psig.

c. FSAR, page 6.1-2

The release of fission products from the containment is limited in three ways:

1. Blocking the potential leakage paths from the containment. This is accomplished by.

a) A steel-lined concrete reactor containment with liner weld channels and double barrier piping penetrations either anchored or utilizing testable expansion bellows which are continuously pressurized to form a virtually leaktight barrier preventing the escape of fission products should a LOCA occur.

b) Isolation of process lines by the containment isolation system and the isolation valve seal water system which imposes water sealed double barriers for selected lines which penetrate the containment.

2. Reducing the fission product concentration in the containment atmosphere. This is accomplished by chemically treated spray which removes elemental iodine vapor from the containment atmosphere by washing action and by recirculation of containment atmosphere through the HEPA filters in the fan cooler units which remove particulate matter.

3. Reducing the containment pressure and thereby limiting the driving potential for fission product leakage. This is accomplished by three independent and redundant containment spray systems of equal heat removal capacity which cool the containment atmosphere and by recirculation of the containment atmosphere through fan cooler units.

Design conditions are not exceeded for as long as accident conditions require the following.

The design pressure (47 psig) and temperature (271° F) of the containment are in excess of the calculated peak pressure (42 psig) and temperature (263° F) occurring as the result of the complete blowdown of the reactor coolant through any rupture of the reactor coolant system (FSAR, page 5.1-4).

The containment structure and all penetrations are designed to withstand the loadings of the design basis accident with the combined design or maximum potential seismic conditions (FSAR, page 5.1-4).

The design pressure is not exceeded during any long-term pressure transient determined by the combined effects of heat sources such as a residual heat and metal-water reactions with minimum operation of the emergency core cooling and the containment fan cooler and spray systems (FSAR, page 5.1-4).

3. Conclusion

The Zion design addresses the criteria contained in GDC 16.

4. References

FSAR Sections

Chapter 5.0 Containment System

6.1.1 Engineered Safety Features Criteria

6.3 Containment Fan Cooler System

6.4 Containment Spray System

6.5 Leakage Detection

6.6 Containment Isolation

Appendix 6A Iodine Removal Effectiveness Evaluation of Containment Spray System

14.2 Standby Safeguards Analysis

14.3 Primary System Pipe Rupture

FSAR Questions

5.1 through 5.84 Containment

11.17 Containment Atmosphere Sampling

11.19 Containment Atmosphere Radiation Monitoring System

13.7 Sump Isolation Valves

14.28 Containment Pressure - Time Response Analysis

SER Sections

3.7.1 Containment Structure

6.1 Containment Systems

STATEMENT OF GDC 17 - ELECTRIC POWER SYSTEMS

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power sources, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling containment integrity, and other vital safety functions are maintained.

Provisions shall be be included to minimize the probability of losing electric power from any of the remaining sources as a result of, or coincident with the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power sources.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 17. Two separate offsite power supplies are provided. Redundant and independent onsite power supplies, both AC and DC, are provided and have appropriate testability.

2. General Review

a. FSAR, Figure 8.4-2

Onsite and offsite electrical systems both are designed with safety functions. This is shown in the single line diagram, FSAR Figure 8.4-2.

b. FSAR, Table 8.4-2

The onsite system can perform safety functions assuming a single failure. The assignment of engineered safety features to the three electrical systems or divisions for each unit is indicated in FSAR Table 8.4-2. The division of the loads between the system buses is such that the total loss of any one of the three electrical systems or divisions on either unit will not prevent the safe shutdown of the reactor under any postulated normal or abnormal condition.

c. FSAR, Question 8.9, and SER, page 8-1

The transmission network supplies electric power by two physically independent circuits. The switchyard at the site is a 345-kV ring bus configuration that provides terminal facilities for six transmission lines. These lines leave the site on two separate rights of way.

d. FSAR, pages 8.1-2 and A-19

Each offsite circuit is designed to be available in sufficient time following loss of all onsite and the other offsite power supplies. Commonwealth Edison's generation and transmission system is designed to withstand the sudden outage of large amounts of capacity. Reliability of electric power supply is insured through independent connections to the system grid and a redundant source of emergency power from five diesel generators installed in the facility. Power to the engineered safety features is assured even with the failure of a single active component in each system.

e. FSAR, Question 8.21

Provisions minimizing the probability of simultaneous loss of all power sources are provided. The probability of such an event is estimated to be extremely low.

3. Conclusion

The Zion design addresses all aspects of Criterion 17.

4. References

FSAR Sections

1.2.7 Electrical Syste"

1.5.7 Emergency Power

Chapter 8 Electrical stems

14.1.8 Loss of External Electrical Load

14.1.12 Loss of All AC Power to the Station Auxiliaries

FSAR Questions

8.1 5000 KVA Standby Diesel Generators

8.2 Standby Diesel Generator Design

8.3 Diesel Generator Experience Clause

8.4 Diesel Starting System

8.5 Diesel Generator Room Design

8.6 Automatic Transfer Switch

8.7 Onsite Standby Power Systems

8.8 345 KV Switchyard Breakers

8.9 345 KV Switchyard and Transmission Facilities

8.10 Buses 111 and 112

8.11 Battery Monitoring

8.12 Seismic Testing of DC Systems

8.13 Circuit Breaker Interlocks

8.14 Circuit Breaker Interlocks

8.15 Auxiliary Power System

8.16 Physical Separation Criteria for Redundant Cables

8.17 Diesel Generator Fuel Oil Systems

8.18 Diesel Generator Trip Circuits

8.20 Automatic Load Dispatching

8.21 Loss of Offsite Power

8.22 4000 KW Standby Diesel Generators

8.23 Diesel Lube Oil System and Jacket Cooling Water System

8.24 Fans and Ducts

8.25 Scram Breaker Cabinet Cabling

8.26 Cable Trays Seismic Design Criteria

<u>SER Sections</u>

Chapter 8.0 Electrical Power

STATEMENT OF GDC 18 - INSPECTION AND TESTING OF ELECTRIC POWER SYSTEMS

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 18. Provisions have been made in the design of periodic inspection and testing of appropriate areas of the systems. Periodic tests can be made of major portions of the power systems under conditions simulating the design conditions.

2. General Review

Electric power systems important to safety shall be designed to permit inspection and testing.

a. FSAR,page 8.1-11

Each diesel generator will be started and loaded for a period of time long enough to bring all components of the system into temperature equilibrium conditions.

b. FSAR, page 8.1-12

The station batteries and other equipment associated with the DC system will be serviced and tested periodically. Periodic testing of all other engineered safety features electrical equipment will be made.

Systems shall be designed to test operability and functional performance of components and systems as a whole, including the protection system and transfer of power.

a. FSAR, Question 8.4

In addition to the monthly testing, each diesel generator will undergo a comprehensive functional test during refueling outages.

b. FSAR, Question 8.11

During every refueling outage, the batteries will be subjected to a rated load discharge test.

c. FSAR, Question 8.15

The automatic transfer of ESS 4-kV buses 147 and 148 and 149 from either the main or reserve feeds to their associated diesel-generators, by deenergizing the 4-kV buses (one at a time), could be performed during normal operations, but such tests are not recommended.

d. FSAR, page 7.2-4

The signal conditioning equipment of each protection channel in service at power is capable of being calibrated and tested independently by simulated analog input signals to verify its operation without tripping the reactor. The testing scheme includes checking through the trip logic to the trip breakers. Thus, the operability of each trip channel can be determined conveniently and without ambiguity. Functional operation of the power sources for the protection system is discussed in Chapter 8.

3. Update to FSAR Review Zion Confirmatory Order

Diesel generator testing is being performed in accordance with Regulatory Guide 1.108.

4. Conclusion

The Zion Station design addresses the criteria contained in GDC 18.

5. References

FSAR Sections

7.2.1 Protection Systems

8.4.4 Tests and Inspections

FSAR Questions

7.12 Reactor Protection System Testing

8.4 Diesel Starting System

8.11 Battery Monitoring

8.12 Seismic Testing of DC systems

8.15 Testability of Auxiliary Power System

SER Sections

7.2 Reactor Protection System

8.3 Onsite Power

Letters

H.R. Denton, NRC, letter to D. L. Peoples, CECO, February 29, 1980, containing Confirmatory Order.

STATEMENT OF GDC 19 - CONTROL ROOM

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 19 except that provisions for cold shutdown from outside the control room have not been incorporated into the plant design. Appropriate radiation protection for the control room and access routes has been provided. Provisions have been made for hot shutdown from outside the control room. Cold shutdown from outside the control room is not contemplated at Zion. The control room has been designed to remain operable and habitable under extremely severe postulated events. Operators will not be forced to leave the control room.

2. General Review

A control room is provided to operate the nuclear unit safely under normal conditions and to maintain it in safe condition under accident conditions, including LOCA's.

a. FSAR, page 7.7-1

The plant is equipped with a control room which contains those controls and instrumentation necessary for operation of the reactor and turbine generator under normal and accident conditions.

b. FSAR, page 7.7-3

The primary objectives in the control room layout are to provide the necessary controls to start, operate, and shutdown the unit with sufficient information display and alarm monitoring to ensure safe and reliable operation under normal and accident conditions.

Adequate radiation protection is provided to permit access and occupancy for the duration of the accident.

a. FSAR, page 7.7-1

Sufficient shielding, distance, and containment integrity are provided to assue that control room personnel shall not be subjected to doses under postulated accident conditions during occupancy of, ingress to and egress from the control room which, in the aggregate, would exceed suggested lim .s in 10 CFR 100.

b. FSAR, Question 9.3

An accumulated thyroid dose for 30 days after a LOCA would be approximately 0.6 rem.

Equipment shall be placed at appropriate locations outside the control room with capability for prompt hot shutdown of the reactor, including instrumentation and controls to maintain unit in safe condition.

a. FSAR, page 7.7-7

The reactor plant can be brought to, and maintained in, a hot shutdown conditon for an extended period of time from outside the control room.

Local control stations are provided for each unit to duplicate controls on the main control panel for those systems required to shut the reactor down to a hot shutdown conditon.

b. FSAR, Question 1.4

The response to FSAR Question 1.4 lists the necessary equipment, systems, and instrumentation that have been provided with local controls and/or readouts to allow the plant to be shutdown and maintained in a hot shutdown condition from outside the control room.

3. Update of FSAR Review

The necessary instrumentation for bringing the plant to cold shutdown from inside the control room will be modified such that the required signals will be independent from the auxiliary electric equipment room and the control room. Modifications will ensure the availability of instrumentation necessary for safe shutdown from outside of the auxiliary electric equipment room or the control room in the event of a fire.

4. Conclusion

The Zion Station de_:_ addresses the criteria of GDC 19.

5. References

FSAR Sections

7.3 Control Systems

7.7 Operating Control Stations

7.7.5 Control Room Availability

7.7.6 Hot Shutdown Control

9.10.3 Control Room HEating, Ventilating and Air Conditioning System

9.10.5 Control Room Ventilation Isolation

11.2.2.2 Control Room Shielding

FSAR Questions

1.4 Shutdown capability

7.13 Control Room Monitors of Primary Plant

7.14 Control Room Indication of Protection System Status

7.27 Control Room Operability of Safety Related Equipment

9.3 Control Room Ventilation System

11.1 Radiation Monitoring System

SER Sections

7.0 Instrumentation and Control Systems

9.6 Air Conditioning and Ventilation Systems

12.1 Shielding

Letters

D.L. Peoples, CECo, letter to H.R. Denton, NRC, April 30, 1980, containing hot shutdown analysis for fire protection.

D.L. Peoples, CECo, leter to H.R. Denton, NRC, May 28, 1980, containing cold shutdown analysis for fire protection.

PROTECTION AND REACTIVITY CONTROL SYSTEMS

STATEMENT OF GDC 20 - PROTECTION SYSTEM FUNCTIONS

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18 December 1971

The Zion Station design conforms with the intent of Criterion 20. The protection system automatically initiates the reactivity control systems as described in Chapter 7 of the FSAR. The system will further sense the postulated accidents and initiate engineered safety features operation.

2. General Review

The protection system is designed to initiate the operation of appropriate systems, including the reactivity control systems to assure specified fuel design limits are not exceeded. It is also designed to sense accident conditions and initiate operation of systems and components important to safety.

a. FSAR, page 7.2-1

If the Reactor Protection System receives signals which are indicative of an approach to unsafe operating conditions, the system actuates alarms, prevents control rod withdrawal, initiates load cutback, and/or opens the reactor trip breakers.

The operating region below these trip settings is designed so that no combination of power, temperatures, and pressure could result in Departure from Nucleate Boiling Ratio (DNBR) less than 1.3 for any credible operational transient with all reactor coolant pumps in operation.

b. FSAR, page 7.5-1

The engineered safety features instrumentation monitors parameters to detect failures in the reactor coolant and steam flow systems and to initiate engineered safety features equipment operation.

3. Conclusion

The Zion Station design addresses the criteria contained in GDC 20.

4. References

FSAR SECTIONS

7.2 Protective Systems

7.5 Engineered Safety Features Instrumentation

14.1 Core and Coolant Boundary Protection Analysis

FSAR Questions

7.2 Protection Systems

7.4 Safety Related Electrical Equipment

7.7 Protection Systems

7.9 Seismic Design Criteria of Reactor Protection Systems

7.11 Reactor Protection System and Engineered Safety Features Actuation System

7.18 Engineered Safety Feature Logic

SER Sections

7.2 Reactor Protection System

7.3 Engineered Safety Features Actuation Systems

STATEMENT OF GDC 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 21 to the maximum extent practicable commensurate with equipment and plant overall safety.

The protection system is comprised of redundant, independent trains of high functional reliability capable of tolerating a single failure. Extensive at-power testing of the systems can be accomplished.

2. General Review

a. FSAR, page 1.3-7

Protection systems are designed with a degree of functional reliability and in-service testability which is commensurate with the safety functions to be performed. The protection systems are designed such that no single failure will prevent proper system action when required.

b. FSAR, page 7.2-2

The protection system design combines redundant sensors and channel independence with coincident trip philosophy so that a safe and reliable system is provided in which a single failure will not violate reactor protection criteria.

c. FSAR, page 1.3-10

The protection system is designed so that removal from service of any component or channel does not result in loss of required redundancy. Bypass removal of a trip circuit is used only in 2/4 logic which then becomes 2/3 logic, except for special 1/2 logic such as start-up trips which become 1/1 logic.

d. FSAR, page 7.2-2

The protection system is designed to permit periodic testing when the reactor is in operation, including capability to test channels independently. Protection channels required for full power operation are designed with sufficient redundancy for individual channel calibration and tests to be made during power operation without degrading the reactor protection.

3. Conclusion

The Zion Station design addresses the criteria contained in GDC 21.

4. References

FSAR Sections

1.3.4 Reliability and Testability of Protection Systems

7.2.1 Protection Systems Reliability

7.2.2 Protection Systems Testing

FSAR Questions

7.11 Protection System Physical Identification

7.12 Protection System Testing

7.14 Protection System Operability Status

SER Sections

7.2 Reactor Protection System

Zion Station Radiological Safety Technical Specifications

3/4.1 Reactor Protection Instrumentation and Logic

STATEMENT OF GDC 22 - PROTECTION SYSTEM INDEPENDENCE

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diverisity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 22. Independent, redundant, and separate subsystems have been provided. Extensive use has been made of diverse sensors for input to the various system functions.

2. General Review

Effects of natural phenomena, normal operation, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of protection function.

a. FSAR, page 7.2-1

For either earthquake (operational or design basis) the equipment is designed to assure that it does not lose its capability to perform its function; i.e., shut the plant down and maintain it in a safe shutdown condition.

b. FSAR, page 7.2-2

Protection channels required for full power operation are designed with sufficient redundancy for individual channel calibration and tests to be made during power operation without degrading the reactor protection.

c. FSAR page 7.2-4

The components of the protection system are designed and arranged so that any adverse environment accompanying an emergency situation in which the components were required to function does not interfere with that function.

d. FSAR, page 7.2-3

Functional diversity or diversity has been used in the design to prevent loss of protection function. The extent of protection system diversity has been evaluated for a wide variety of postulated accidents. Generally, two or more diverse protection functions would terminate an accident before intolerable consequences could occur. The design approach...provides a protection system which continually monitors numerous system variables by different means; i.e., protection system diversity.

3. Conclusion

The Zion Station design addresses the criteria in GDC 22.

4. References

FSAR Sections

7.2.1 Protection Systems Redundancy and Independence

7.2.2 Protective System Independence

FSAR Questions

7.9 Seismic Design Criteria

7.10 Redundant Reactor Protection Systems

SER Sections

7.2 Reactor Protection System

STATEMENT OF GDC 23 - PROTECTION SYSTEM FAILURE MODES

The protection system shall be designed to fall into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire pressure, steam, water, and radiation) are experienced.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 23. Those portions of the protection system which are not "fail safe" are either protected against the failure mode of concern or are backed up by functionally redundant, physically separate systems or both.

2. General Review

The protection system is designed to fall into safe or acceptable state if disconnection of the system, loss of energy, or postulated adverse environments are experienced.

a. FSAR, page 7.2-4

Each reactor trip channel is designed on the "deenergize to operate" principle; a loss of power causes that channel to go into its trip mode. All safety-related, air-operated valves are spring loaded to move to the "fail-safe" position on loss of instrument air... The entire protection system is thus inherently safe in the event of a loss of power.

b. FSAR, page 7.2-10

The protective channels are designed to perform their functions when subjected to adverse environmental conditions.

3. Conclusion

The Zion Station design addresses the criteria contained in GDC 23.

4. References

FSAR Sections

7.2.1 Protection Systems

FSAR Questions

7.14 Protection System Available

7.23 Fuel Handling System Interlocks Failure Criteria

7.27 Environmental Conditions

7.28 Radiation Exposure of Electrical and Mechanical Equipment of Reactor Protection System

7.29 Environmental Conditions

SER Sections

7.2 Reactor Protection System

Zion Station Radiological Safety Technical Specifications

3/4.1 Reactor Protection Instrumentation and Logic

STATEMENT OF GDC 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS

Separation of protection and control systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 24. Failure of, or removal from service of a component or channel of either the protection system or the control system will not impair plant safety. The redundant and diverse nature of the protection and control systems assures that such a single failure will not negate safety functions. The use of isolation amplifiers between protection and control signal paths offers further assurance of continued safety.

2. General Review

The protection system is designed so that it is separated from control systems.

a. FSAR, page 7.2-3

In the control and protection systems, the control system is separate and distinct from the protection system. Although the protection system is independent of the control system, the control system is dependent upon signals derived from the protection system through isolation amplifiers.

b. FSAR, page 7.2-6

Where protection signal intelligence is required for other than protection functions, an isolation amplifier (part of the protection set) is used to transmit the intelligence. The isolation amplifier prevents the perturbation of the protection channel signal (input) due to any disturbance of the isolated signal (output) which normally could occur near any termination of the output wiring external to the protection racks.

3. Conclusion

The Zion Station design addresses the criteria contained in GDC 24.

4. References

FSAR Sections

7.2.1 Protection Systems Redundancy and Independence

7.2.2 Protective System Independence

FSAR Questions

7.1 RCS Loop Interlock Assurance

7.10 Separation of Protection Systems

7.11 Identification of Protection and Engineered Safety Systems

SER Sections

7.2 Reactor Protection System

7.7 Control Systems

STATEMENT OF GDC 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 25. The reactivity control systems are such that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

2. General Review

The protection system is designed to assure that acceptable fuel limits are not exceeded for a single malfunction of reactivity control systems.

a. FSAR, page 7.2-5

Reactor shutdown with RCCA's is completely independent of the normal control functions since the trip breakers interrupt the power to the full length rod mechanisms regardless of existing control signals.

b. FSAR, page 3.1.2-7

The reactor protection systems are capable of protecting against any single credible malfunction of the reactivity control system, by limiting reactivity transients to avoid exceeding acceptable fuel design limits.

3. Conclusion

The Zion Station design addresses the criteria contained in GDC 25.

4. References

FSAR Sections

3.1.2 Reactivity Control Systems Malfunction

7.2.1 Protection Systems

7.2.2 Reactor Protection System Description

14.1.2 Uncontrolled RCC Assemble Withdrawal at Power

FSAR Questions

7.2 RCS Loop Protection

7.7 Reactor Trip Actuation

7.8 Control Systems

SER Sections

7.2 Reactor Protection System

7.7 Control Systems

STATEMENT OF GDC 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 26. Two independent reactivity control systems, rod control clusters and soluble boric acid, are provided and have the capabilities for shutdown discussed in the criterion, except that the boric acid system is not designed to be a means of compensating for rapid reactivity transients resulting from operations such as load following.

2. General Review

a. FSAR, page 3.1.2-4

Two independent reactivity control systems are provided, one involving rod cluster control (RCC) assemblies and the other involving chemical shimming.

b. FSAR, page 3.1-2-5

One system is designed so that by using control rods it is capable of controlling reactivity changes to assure under normal operation, including anticipated operational occurrences, and with the appropriate margin for stuck rods, specified acceptable fuel design limits are not exceeded.

The reactor core, together with the reactor control and protection system is designed so the minimum allowable DNBR is at least 1.30 and there is no fuel melting during normal operation, including anticipated transients.

The shutdown groups are provided to supplement the control groups of RCC assemblies to make the reactor at least one per cent subcritical at the hot zero power condition (K_{eff}=0.99) following trip from any credible operating condition, assuming the most reactive RCC assembly is in the fully withdrawn position.

c. FSAR, page 7.3-3

The second reactivity system is designed so it is capable of controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout).

The reactor control system will also initially compensate for reactivity changes caused by fuel depletion and/or xenon transients. Long-term compensation for these two effects is periodically made by adjustments of the boron concentration to return the rod control bank to its normal operating range.

d. FSAR, pages 9.2-3 and 3.1.2-6

One of the systems is designed so it is capable of holding the core subcritical under cold conditions. Shutdown for long-term and reduced temperature conditions can be accomplished with boric acid injection using redundant components. Any time that the reactor is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection always exceeds that quantity required for the normal cold shutdown.

3. Conclusion

The Zion Station design addresses the criteria contained in GDC 26.

4. References

FSAR Sections

3.1.2 Redundancy of Reactivity Control and Capability

7.2.1 Redundancy of Reactivity Control

9.2.1 Redundancy of Reactivity Control and Capability

FSAR Questions

3.3 Central Control Rod Missing

7.8 Control Systems

7.15 Boric Acid Subsystem

9.7 Chemical Volume and Control System (CVCS)

<u>SER Sections</u>

7.7 Control Systems

9.2 Chemical and Volume Control System (CVCS)

STATEMENT OF GDC 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10 Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 27. Appropriate reactivity margin is available under postulated accident conditions to assure that the capability to cool the core is maintained. This margin includes an allowance for the most reactive rod control cluster being stuck out of the core.

2. General Review

The reactivity control systems are designed with the capability to control reactivity reliably during an accident condition.

a. FSAR, page 3.1.2-5

Normal reactivity shutdown capability is provided 2 seconds following a trip signal by control rods with boric acid injection used to compensate for the long-term decay transient and for plant cooldown.

b. FSAR, page 9.2-2

Anytime that the plant is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection always exceeds the quantity required for the normal cold shutdown. This quantity always exceeds the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

3. Conclusion

The Zion Station design addresses the criteria contained in GDC 27.

4. References

FSAR Sections

3.1.2 Principle Design Criteria

3.1.3 Safety Limits - Reactivity Control Limits

3.2.1 Nuclear Design and Evaluation

7.2 Protective Systems

7.3 Control Systems

9.2 Chemical and Volume Control System

FSAR Questions

3.7, 3.1 Withdrawal of Single Control Rod

3.3 Central Control Rod Missing

3.8 Violation of (a) Design Enthalpy Rise Peaking Factor and (b) Safety Limit

7.7 Protection System (which actuates reactor trip and ESF action)

7.8 Control Systems

7.12 Reactor Protection System Testing

SER Sections

4.3 Nuclear Design

7.2 Reactor Protection System

9.2 Chemical and Volume Control System

STATEMENT OF GDC 28 - REACTIVITY LIMITS

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 28. The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing both control rods and boron removal are limited to values which prevent rupture of the coolant pressure boundary or disruption of the core or internals to a degree which could impair the effectiveness of ECCS. Rod control cluster ejection and other postulated accidents have been considered.

2. General Review

Reactivity controls are designed to set limits on potential amount and rate of reactivity increase.

Reactivity Control Via Rod Control Cluster (RCC)

a. FSAR page 3.1.1-1

RCC's are employed to terminate any credible power transient. This termination can occur with the most reactive control cluster stuck in the fully withdrawn position.

b. FSAR, page 3.1.2-4

RCC assemblies can hold the reactor subcritical from any mode of operation.

c. FSAR, pages 3.1.2-5 and 7.2-19

Normal reactivity shutdown capability is provided within 2 seconds following a reactor trip signal by freefall of the full length RCC's.

Reactivity Control Via Chemical Shimming

d. FSAR, pages 3.1.2-6 and 9.2-2

Boric acid solution maintains the shutdown margin of reactivity over extended periods of time. Redundant paths for adding boric acid are available. In addition there is more boric acid in the boric acid storage tanks than is required for cold shutdown.

Reactivity control systems are designed so that they are capable of preventing damage to reactor coolant pressure boundary and loss of core cooling capability.

a. FSAR, page 7.2-1

If the reactor protection system receives signals which are indicative of an approach to unsafe operating conditions, the system actuates alarms, prevents control rod withdrawal, initiates load cutback, and/or opens the reactor trip breakers.

b. FSAR, page 3.1.3-5

Adequate clearance is provided between the absorber rods and guide thimbles so that coolant flow is sufficient to remove the heat generated.

Control rod drive assemblies are hermetically sealed to prevent reactor coolant leakage.

For reactivity accident analysis, the following phenomena are considered:

a. FSAR, page 7.2-2

Rod stops from nuclear overpower, overpower ΔT, and overtemperature ΔT deviation are provided to prevent abnormal power conditions which could result from excessive control rod withdrawal initiated by a malfunction of the reactor control system or by operator violation of administrative procedures.

b. FSAR, page 6.2-1

Following a double-ended serverence of a steam line an overpower reactor trip would occur.

c. FSAR, page 7.2-1

The basic reactor operating philosophy is to define an allowable region of power, pressure, and coolant temperature conditions. This allowable range is defined by the primary tripping function: The overpower ΔT trip and the nuclear overpower trip.

d. FSAR, page Q14.12-4

Any reactivity insertion due to cold water addition from an isolated loop would be more severe at the hot shutdown condition since the value of the moderator temperature coefficient of reactivity is a decreasing function of temperature. Accordingly, the transient associated with return of an isolated loop to service has been analyzed for the plant in the hot shutdown condition.

e. Zion Station Radiological Safety Technical Specifications, page 76

The hot leg stop valve and the cold leg stop valve shall not be opened unless the boron concentration of the isolated loop is greater than or equal to the boron concentration in the unisolated loops.

f. FSAR, page 14.2.6-2

In the event of a rupture of the control rod mechanism housing resulting in a control rod ejection, the operation of a chemical shim plant is such that the severity of this accident is inherently limited. Boron is used to control reactivity and most control rods are fully withdrawn from the core. Therefore, should a control rod be ejected, there would probably be no reactivity excursion. Occasionally, it may be desirable to operate with larger than normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the control rods above this limit guarantees adequate shutdown capability and acceptable power distribution.

3. Conclusion

The Zion Station design addresses the criteria in GDC 28.

4. References

FSAR Sections

3.1.2 Reactivity: Design Criteria

3.1.3 Safety Limits

3.2.1 Reactivity Coefficient

7.2.1 and 9.2.1 Design Bases

14.1 - Core Protection Analysis

14.2.6 Rupture of a Control Rod Drive Mechanism Housing (RCC Assembly Ejection)

FSAR Questions

3.1 Consequences of Rod Withdrawal

3.3 Control Rod Insertion Limit

14.12 Startup of an Inactive Reactor Coolant Loop

Zion Station Radiological Safety Technical Specifications

3.3.1 (Reactor Coolant System) Operation Components

STATEMENT OF GDC 29 - PROTECTION AGAINST ANTICIPATED OPERATIONAL OCCURRENCES

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

EVALUATION OF COMPLIANCE

FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 29. The protection and reactivity control systems are designed to insure an extremely high probability of fulfilling their intended functions. The design principles of diversity and redundancy coupled with a rigorous quality assurance program support this probability as does experience in operating plants using the same basic design.

2. General Review

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

a. FSAR, page 7.2-3

The protection and control systems are separate and identifiable. The design approach permits not only redundancy of control, providing its own desirable increment to overall plant safety, but also provides a protection system which continuously monitors numerous system variables by different means; i.e., protection system diversity.

Required continuous power supply for the protection systems is discussed in Chapter 8.

b. FSAR, page 9.1-1

As described in Chapter 7 and justified in Chapter 14, the reactor protection systems are designed to limit reactivity to transients to DNBR ≥1.3 due to any single malfunction in the reactivity control system.

3. Conclusion

The Zion Station design addresses the criteria set forth in GDC 29.

4. <u>References</u>

<u>FSAR Sections</u>

3.1.2 Reactor Core Design

3.2.1.1 Reactivity Control Aspect

3.2.3.4 Evaluation of Core Components

7.2.1 Protection Systems

7.2.2 Reactor Protection System and Protective Actions

7.2.3 Specific Control and Protection Interactions

8.4.1.3 and 8.5 120 VAC Instrumentation and Control Power

9.1.2 Reactivity Control Systems Malfunction

13.4 Operation Restrictions

<u>FSAR Questions</u>

7.2 Reactor Control Protection System Adjustments

7.7 Reactor Protection System Equipment List

FLUID SYSTEMS

STATEMENT OF GDC 30 - QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 30. The quality levels employed for the reactor coolant pressure boundary are extremely comprehensive. Systems have been included in the plant to detect and to the extent practical, to locate leaks.

2. General Review

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical.

a. FSAR, page Q4.1-1

Suppliers' capabilities are evaluated to assure that they are able to manufacture quality materials. All specifications used in procurement are identical to the specifications utilized for domestic procurement. Resident quality assurance is maintained to assure compliance with specifications.

b. FSAR, page 4.1-23

All primary pressure-containing components of the reactor coolant system are designed, fabricated, inspected, and tested in conformance with the applicable codes and are Class I seismic design.

c. FSAR, page 4.5-1

Table 4.5-1 summarizes the quality assurance program for all reactor coolant components. In this table, all of the nondestructive tests and inspections which are required by Westinghouse specifications on reactor coolant system components and materials are specified for each component. All tests required by the applicable codes are included in this table. Westinghouse requirements, which are more stringent in some areas than those requirements specified in the applicable codes, are also included. The fabrication and quality control techniques used in the fabrication of the reactor coolant system were equivalent to those used for the reactor vessel.

d. FSAR, page 4.5-12

The ASME Code for Inservice Inspection of Nuclear Reactor Coolant System is being used as a general guideline for in-service inspection requirements to the maximum extent possible.

Means shall be provided for detecting and to the extent practical, identifying the location of the source of reactor coolant leakage.

Provisions have been made in the design and arrangement of the reactor coolant system, engineered safety systems and certain associated auxiliary systems to allow access for in-service inspection.

With regard to the reactor coolant system components, the layout of the equipment and support structures is designed to permit access to the following for examination during a plant shutdown. Access implies ability to visually examine surfaces and perform other required examinations (FSAR, page 4.5-12).

3. Conclusion

The Zion Station design addresses the criteria of GDC 30.

4. References

FSAR Sections

4.1.7 Codes and Classifications

4.5.1.1 Non-Destructive Inspection of Material and Components

4.5.1.2 In-Service Inspection Capability

FSAR Questions

4.1 Foreign Suppliers

4.10 Inservice Inspection

4.14 RCS Component Codes

4.38 through 4.41 RCS Leakage

6.7 Valve Leaks

6.10 Inservice Inspection of Fluid Systems

Zion Station Radiological Safety Technical Specifications

3/4.3.4 (Reactor Coolant System) Structural Integrity

STATEMENT OF GDC 31 - FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 31. The reactor coolant pressure boundary is designed so that, for all normal operating and postulated accident modes, the boundary behaves in a nonbrittle manner and so that the probability of rapidly propagating failure is minimized. Service temperature and pressure; irradiation, cyclic loading; seismic, blowdown and thermal forces from postulated accidents, residual stresses, and code allowable material discontinuities have all been considered in the design with appropriate margins for each.

2. General Review

   a. FSAR, page 4.1-10

   The reactor coolant pressure boundary is designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary behaves in a non-brittle manner. Since the normal operating temperature of the reactor vessel is well above the maximum expected DTT, brittle fracture during normal operation is not considered to be a credible mode of failure.

   b. FSAR, page 4.1-8

   The reactor coolant pressure boundary is designed to reduce to an acceptable level the probability of a rapidly propagating type failure.

   c. FSAR, pages 4.1-11 and 4.1-23

   The design temperature for each component is selected to be above the maximum coolant temperature in that component under all normal and anticipated transient load conditions. Reactor vessel design is based on the transition temperature method of evaluating the possibility of brittle fracture of the vessel material as a result of operation.

d. FSAR, pages 4.1-5 and 4.2-25

The reactor coolant system in conjunction with its control and protective provisions is designed to accomodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions, and maintain the stresses within applicable code stress limits.

The materials of construction of the pressure retaining boundary are protected by control of coolant chemistry from corrosion phenomena which might otherwise reduce the system structural integrity during its service lifetime.

The nil-ductility transition (NDT) temperature of the reactor vessel material opposite the core is established at a Charpy V-notch test value of 30 ft-lb. or greater. The material is tested to verify conformity to specified requirements and to determine the actual NDT temperature value. In addition, this material is 100 percent volumetrically inspected by ultrasonic tests using both straight beam and angle beam methods.

The remaining material in the reactor vessel and other reactor coolant system components meets the appropriate design code requirements and specific component function.

e. FSAR, page 4.5-1

As a result of fast neutron irradiation in the region of the core, the material properties will change, including an increase in the NDT temperature of +10° F for Unit 1 and +35° F for Unit 2 in this region has been established during fabrication. In the surveillance programs, the evaluation of radiation damage is based on preirradiation testing of Charpy V-notch and tensile specimens and postirradiation testing of Charpy V-notch, tensile, and wedge opening loading (WOL) fracture mechanics specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approved and the fracture mechanics approach. These programs are in accordance with ASTM-E-185-70, "Recommend Practice for Surveillance Tests for Nuclear Reactor Vessels."

f. FSAR, page B-1

The normal as well as abnormal loads for vessels and piping are considered singly and in combination (see Table B1-1), and the allowable stress limits for each of the possible combinations are limited to those specified in Table B1-2.

g. FSAR, page 4.5-6

Procedures for performing the examinations were consistent with those established in the ASTM Code Section III and were reviewed by qualified engineers. These procedures have been developed to provide the highest assurance of quality material and fabrication. Not only are the size of flaws considered, but also how the material is fabricated, the orientation and type of possible flaws, and the areas of most severe service conditions.

3. Conclusion

The Zion Station design addresses the criteria of GDC 31.

4. References

FSAR Sections

4.1.3 Principal Design Criteria

4.1.4 Design Characteristics

4.1.6 Service Life

4.5.1 Reactor Coolant System Inspection

FSAR Questions

4.2.2 Reactor Coolant System Boundary Materials Testing

Zion Station Radiological Safety Technical Specifications

3.3 Reactor Coolant System

3.3.4 Structural Integrity

STATEMENT OF GDC 32 - INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 32. The Zion Station reactor coolant pressure boundary will be periodically inspected under the provisions of ASME Section XI. A reactor vessel metal surveillance program will be employed.

2. General Review

   a. FSAR, page 4.1-9

   The design of the reactor vessel and its arrangement in the system provides the capability for accessibility during service life to the entire internal surfaces of the vessel and certain external zones of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads. The reactor arrangement within the containment provides sufficient space for inspection of the external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete.

   b. FSAR, page 4.1-9 and 4.10

   An appropriate material surveillance program is conducted for the reactor pressure vessel. Samples of reactor vessel plate materials are retained and catalogued in case future engineering development shows the need for further testing. The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens.

3. Conclusion

The Zion Station design addresses the criteria of GDC 32.

4. References

FSAR Sections

4.1.3 Reactor Coolant Pressure Boundary Surveillance

FSAR Questions

4.4 Reactor Vessel Material Surveillance Program

Zion Station Radiological Safety Technical Specifications

3/4.3.4 (Reactor Coolant System) Structural Integrity

STATEMENT OF GDC 33 - REACTOR COOLANT MAKEUP

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

EVALUATION OF COMPLIANCE

1. FSAR Qustion 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 33. The normal flow path for reactor coolant system charging can be used to assure appropriate makeup supply for small breaks.

2. General Review

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided.

a. FSAR, page 14D-1

Should a loss of coolant occur, depressurization of the reactor coolant system causes fluid to flow to the reactor coolant system from the pressurizer... The safety injection system is actuated when the pressurizer low pressure and low level setpoints are reached...safety injection system actuation is also provided by a high containment pressure signal. Note that as a result of the Three Mile Island incident pressurizer level was deleted from safeguard logic per Zion Amendment Nos. 49 and 46.

b. FSAR, page 6.2-7

The active components...provide injection for small break ruptures where the primary coolant pressure does not drop below the accumulator pressure for an extended period of time after the accident.

Electric power sources shall be provided for safe shutdown and offsite power generation.

In the event of total loss of auxiliary power from off-site sources, diesel generators will supply the electric power required for safe shutdown. Commonwealth Edison also has an internal electric power reserve of over 1847 megawatts (FSAR, page 8.1-2).

For off-site electric power requirements Zion Station is interconnected with one Wisconsin substation and three other substations in Illinois (FSAR, page 8.1-2).

3. Conclusion

The Zion Station design addresses the criteria in GDC 33.

4. References

FSAR Sections

6.2.2 (ECCS) System Design and Operation

8.2 345 kV Network Transmission Terminal

9.2 Chemical Volume and Control System

14.3 Reactor Coolant System Pipe Rupture

Appendix 14D Effect of Revised Safety Injection System

Letters

A. Schwarz, NRC, letter to C. Reed, CECo, May 3, 1979 (issuing License Amendment Nos. 49 and 46).

STATEMENT OF GDC 34 - RESIDUAL HEAT REMOVAL

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 34. The residual heat removal system, consisting of two redundant trains of pumps and heat exchangers, has appropriate heat removal capacity to ensure fuel protection. This system supplements the normal steam and power conversion system which is used for first stage cooldown. The auxiliary feedwater system complements the steam and power conversion system in this function. The systems together accommodate the single failure criteria.

2. General Review

a. FSAR, page 9.4-1

The residual heat removal system is designed to remove residual and sensible heat from the core and reduce the temperature of the reactor coolant system during the second phase of plant cooldown.

The system safety function is designed to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. The residual heat removal system provides sufficient capability in the emergency operational mode to accommodate any single active/passive failure and still function in a manner to avoid risk to the health and safety of the public.

b. FSAR, page 9.4-2

For suitable redundancy, the residual heat removal system consists of two residual heat exchangers and two residual heat removal pumps and associated piping, valves, and instrumentation.

c. FSAR, page 9.4-5

Since the residual heat removal system is required for long-term, postaccident removal of decay heat from the reactor core and containment, independent piping systems are provided for the redundant active components so that excessive leakage resulting from the deterioration of, or failure in, some passive element in the system can be identified and isolated without complete system loss-of-function.

d. FSAR, page 9.4-4

Isolation capabilities are provided to assure that for onsite electric power system operation and for offsite electric power system operation the system safety function can be accomplished, assuming a single failure. Isolation of the residual heat removal system is achieved with two remotely operated series stop valves in the pipe from the reactor coolant system to the residual heat removal pump suction and by two check valves in series plus a remotely operated stop valve in each line from the residual heat removal pump discharge to the reactor coolant system.

3. Conclusion

The Zion Station design addresses the criteria of GDC 34.

4. References

FSAR Sections

9.4.1 Design Bases - Codes and Classifications

9.4.2 System Design and Operation

9.4.3 System Design Evaluation - Incident Control

FSAR Questions

6.16 NPSH of ECCI

6.18 Long Term Post-LOCA Cooling

Zion Station Radiological Safety Technical Specifications

3.3.4, 4.3.4 Structural Integrity

STATEMENT OF GDC 35 - EMERGENCY CORE COOLING

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 35. Appropriate core cooling systems have been designed so as to provide for the removal of core thermal loads and for the limiting of metal water reactions to an insignificant level. Suitable redundancy is provided in core cooling systems. The charging/safety injection, accumulator, and safety injection systems will accommodate a single active failure and still fulfill their intended safety function. The residual heat removal system will accommodate a single passive or active failure and still fulfill its intended safety function.

2. General Review

A system to provide abundant emergency core cooling shall be provided.

a. FSAR, page 6.2-1

The emergency core cooling system can afford protection for the following:

1. All pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop, assuming unobstructed discharge from both ends,

2. A loss of coolant associated with the rod ejection accident, and

3. A steam generator tube rupture.

b. <u>Amendment No. 53 to Facility Operating License No. DPR-34 and Amendment No. 50 to Facility Operating License No. DPR-48</u>

Amendment No. 53 to Facility Operating License No. DPR-34 and Amendment No. 50 to Facility Operating License No. DPR-48, for Zion Station Units 1 and 2 update the analysis of the emergency core cooling system. This amendment was approved by the NRC as indicated in the conclusion of the Safety Evaluation Report which states that:

"Based on the review of the submitted documents, we conclude that the results of the LOCA analysis performed with FQ=1.93 are conservative relative to the 10 CFR 50.46 criteria. We consider the resulting changes to the Technical Specifications acceptable for operating Units 1 and 2 with a maximum of 1 percent of the steam generator tubes plugged."

The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant as discussed above. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling, and before the cladding is so embrittled as to fall during or after quenching (FSAR, page 6.2-2). The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of cladding in the reactor (FSAR, page 6.2-2).

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

a. <u>FSAR, page 6.2-2a</u>

Redundancy, diversity, and segregation of instrumentation and components are incorporated to assure that postulated malfunctions will not impair the ability of the emergency core cooling system to meet the design criteria. The system is effective in the event of loss of normal plant auxiliary power coincident with a LOCA, and can accommodate the failure of any single component or instrument channel to respond actively in the system.

b. <u>FSAR, page 6.2-33</u>

Separate single failure analyses were performed for both the injection and recirculation phases of an accident. Tables 6.2-6 and 6.2-7 summarize the results of the single failure analyses.

3. Conclusions

The Zion Station design addresses the criteria of GDC 35.

4. References

FSAR Sections

6.2 Emergency Core Cooling System

14.3 Reactor Coolant System Pipe Rupture

Amendments

Amendment Nos. 53 and 50 to Facility Operating License Nos. DPR-39 and DPR-48 for Zion Station Units 1 and 2

STATEMENT OF GDC 36 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design provides for inspection of the emergency core cooling branch line connections to the reactor coolant system in accordance with the provisions of ASME Section XI. These are the areas of principal stress in the system due to temperature gradients. The remainder of the systems will be verified as to integrity and functioning by means of periodic testing as described in the technical specifications. On this basis, the Zion Station design conforms to the intent of Criterion 36.

2. General Review

   a. FSAR, page 6.2-3

   Design provisions are made to facilitate access to the critical parts of the reactor internals, injection nozzles, pipes, valves, and ECCS pumps for visual or boroscopic inspection for erosion, corrosion and vibration wear evidence and for non-destructive inspection where such techniques are desirable and appropriate.

   b. FSAR, page 6.2-36

   All active and passive components of the emergency core cooling system are inspected periodically to demonstrate system readiness.

   c. Zion Station Radiological Safety Technical Specifications, Section 4.8

   The surveillance requirements for the emergency core cooling system are given in Section 4.8 of the Zion Station Radiological Safety Technical Specifications.

3. Conclusions

The Zion Station design addresses the criteria of GDC 36.

4. References

FSAR Sections

6.2 Emergency Core Cooling System

14.3 Reactor Coolant System Pipe Rupture

FSAR Questions

6.10 Inservice Inspection

Zion Station Radiological Safety Technical Specifications

4.8 Emergency Core Cooling System Surveillance Requirements

STATEMENT OF GDC 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM

Testing of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms to the intent of Criterion 37. Periodic tests will demonstrate the integrity, operability, and performance of the system. The system as a whole, and the entire operational sequence of actuation, power transfer, and cooling water operation will be tested in several phases rather than in one phase during periodic testing.

2. General Review

a. FSAR, page 6.2-37

Each active component of the emergency core cooling system may be actuated on the normal power source at any time during plant operation to demonstrate operability.

b. Zion Station Radiological Safety Technical Specifications, Section 4.8

The testing requirements for the emergency core cooling system are given in Section 4.8 of the Zion Station Radiological Safety Technical Specifications.

3. Conclusions

The Zion Station design addresses the criteria of GDC 37.

4. References

FSAR Sections

6.2 Emergency Core Cooling System

14.3 Reactor Coolant System Pipe Rupture

<u>FSAR Questions</u>

6.10 Inservice Inspection

<u>Zion Station Radiological Safety Technical Specifications</u>

4.8 Emergency Core Cooling System Surveillance Requirements

STATEMENT OF GDC 38 - CONTAINMENT HEAT REMOVAL

Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptable low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with and goes beyond the intent of Criterion 38. Two diverse heat removal systems, each composed of redundant components, are provided. These are the containment spray system (three 100% capacity pumping systems) and the containment fan cooler systems (five units provided, three required for accident heat removal).

2. General Review

A system to remove heat from the reactor containment shall be provided.

a. FSAR, page 6.3-1

The containment fan cooler system is designed to filter, cool, and dehumidify the reactor containment during both normal and abnormal conditions.

b. FSAR, page 6A-1

The containment spray system is an engineered safety system employed to reduce pressure and temperature in the containment following a postulated LOCA.

The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

a. FSAR, page 6.3-5

The reactor containment fan cooler system provides the design heat removal capacity for the containment following a loss-of-coolant accident assuming that the core residual heat is released to the containment as steam.

b. FSAR, Section 14.3.4

The containment heat removal capability of the containment heat removal systems is discussed in FSAR Section 14.3.4.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) its safety function can be accomplished assuming a single failure.

a. FSAR, page 6.3-6

A failure analysis has been made on all active components of the containment fan cooler system to show that the failure of any single active component will not prevent fulfilling the design function. Electrical power to the fan coolers is supplied from the emergency diesel generators upon loss of auxiliary power.

b. FSAR, page 6.4-1

The containment spray system has been divided into three independent 100% capacity subsystems. A single active or passive failure in any of the three subsystems will not affect the operation of either of the other two subsystems. Of the three containment spray pumps, two are motor-driven and the third is diesel-engine driven.

c. FSAR, page 6.4-2

Both motor-driven containment spray pumps and all motor operated valves can be supplied with power from the emergency diesel generators in the event of a loss of offsite power. Failure of a single diesel or emergency bus will affect one subsystem only.

3. Conclusion

The Zion Station design addresses the criteria contained in GDC 38.

4. References

FSAR Sections

6.3 Containment Fan Cooler System

6.4 Containment Spray System

Appendix 6A Iodine Removal Effectiveness

14.3.4 Containment Integrity Evaluation

FSAR Questions

6.6 Fan Cooler Proof Testing

6.14 Fan Cooler Motor Testing

6.15 Fan Cooler Damper Operation

6.16 Spray Pump NPSH

Zion Radiological Safety Technical Specifications

3/4.5 Reactor Containment Fan Coolers

3/4.6 Containment Spray

STATEMENT OF GDC 39 - INSPECTION OF CONTAINMENT HEAR REMOVAL SYSTEM

Inspection of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 39, except for limited areas. The fan cooler units are continually operating (four out of five) and are rotated in service to provide continuous verification of operability and integrity. Access for routine maintenance and inspections has been provided.

The containment spray system integrity will be verified by means of periodic testing as described in the technical specifications. Access has been provided for routine maintenance and inspections, except for the spray ring headers. No provision has been made for inspecting the spray ring headers and nozzles. Provisions have been made to smoke test these headers and nozzles periodically.

2. Updated Reponse to FSAR Question 1.10

Hot air tests using infrared photo equipment have been substituted for smoke tests to verify that the nozzles on the spray ring headers are not restricted.

3. General Review

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

a. Zion Station Radiological Safety Technical Specifications, Section 4.5

The surveillance requirements for the containment fan coolers are given in Section 4.5 of the Zion Station Radiological Safety Technical Specifications.

b. Zion Station Radiological Safety Technical Specifications, Section 4.6

The surveillance requirements for the containment spray system are given in Section 4.6 of the Zion Station Radiological Safety Technical Specifications.

4. Conclusion

The Zion Station design addresses the criteria of GDC 39.

5. References

FSAR Sections

6.3 Containment Fan Cooler System

6.4 Containment Spray System

FSAR Questions

6.10 Inservice Inspection

Zion Station Radiological Safety Technical Specifications

4.5 Reactor Containment Fan Coolers Surveillance Requirements

4.6 Containment Spray Surveillance Requirements

STATEMENT OF GDC 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

Testing of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms to the intent of Criterion 40. See response to Criterion 37.

2. General Review

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing as detailed above.

a. Zion Station Radiological Safety Technical Specifications, Section 4.5

The testing requirements for the containment fan cooler system are given in Section 4.5 of the Zion Station Radiological Safety Technical Specifications.

b. Zion Station Radiological Safety Technical Specifications, Section 4.6

The testing requirements for the containment spray system are given in Section 4.6 of the Zion Station Radiological Safety Technical Specifications.

3. Conclusions

The Zion Station design addresses the criteria of GDC 40.

4. References

FSAR Sections

6.3 Containment Fan Coolers

6.4 Containment Spray System

FSAR Questions

6.8 Containment Fan Cooler Tests

6.10 Inservice Inspection

6.14 Containment Fan Cooler Motor Tests

Zion Radiological Safety Technical Specifications

4.5 Reactor Containment Fan Coolers Surveillance Requirements

4.6 Containment Spray Surveillance Requirements

STATEMENT OF GDC 41 - CONTAINMENT ATMOSPHERE CLEANUP

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 41. The containment spray system, containment fan coolers, and containment purge (Post LOCA) provide controls on and means for reduction of fission products and other substances.

2. General Review

Systems to control fission products, hydrogen, oxygen, and other substances that may be released into the reactor containments shall be provided as discussed above.

a. FSAR, page 6.3-1

The containment fan cooler system is designed to filter, cool, and dehumidify the reactor containment environment during both normal and abnormal conditions. For postaccident operation, the equipment is designed for $2x10^6$ rads during the first 3 hours and an integrated dose of $2x10^8$ rads. Under post-accident conditions, the moisture eliminators are designed to remove not less than 95% of the free water particles 10 microns and greater in size with 0.35 lb entrained water per 1000 cfm.

b. FSAR, page 6.4-1

The containment spray system is designed to limit the pressure in the containment atmosphere to a level below the design pressure and to remove sufficient iodine from the containment atmosphere to limit the offsite boundary doses to values below those set by 10 CFR 100 in the unlikely event of a LOCA.

c. FSAR, Sections 6.8 and 14.3.6

Two modes of hydrogen gas control are available at Zion: (1) Hydrogen Recombiners and (2) Containment Purge System.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electrical power sytem operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) its safety function can be accomplished assuming a single failure.

a. FSAR, page 6.3-6

A failure analysis has been made on all active components of the containment fan cooler system to show that the failure of any single active component will not prevent fulfillment of the design function. Electrical power to the fan coolers is supplied from the emergency diesel generators upon loss of auxiliary power.

b. FSAR, page 6.4-1

The containment spray system has been divided into three independent 100% capacity subsystems. A single active or passive failure in any of the three subsystems will not affect the operation of either of the other two subsystems. Of the three containment spray pumps, two are motor-driven and the third is diesel-engine driven.

c. FSAR, page 6.4-2

Both motor-driven containment spray pumps and all motor-operated valves can be supplied with power from the emergency diesel generators in the event of a loss of offsite power. Failure of a single diesel or emergency bus will affect one subsystem only.

d. FSAR, Question 14.2

Analysis shows that hydrogen concentration inside containment following a loss-of-coolant accident will not reach the lower flammability limit for 37 days. Containment purging can begin 30 days after an accident without exceeding 10 CFR 100 limits. Therefore, any loss of electrical power can be rectified prior to needing a Hydrogen Control System.

e. FSAR, Sections 6.8 and 14.3.6

Two modes of hydrogen gas control are available at Zion: (1) hydrogen recombiners and (2) the containment purge system.

3. Conclusion

The Zion Station design addresses the criteria contained in GDC 41.

4. References

FSAR Sections

6.3	Containment Fan Cooler System
6.4	Containment Spray System
6.8	Recombiner
6A	Iodine Removal Effectiveness
14.3.4	Containment Integrity Evaluation
14.3.5	Environmental Consequences of a LOCA
14.3.6	Controlled Containment Venting After a LOCA

FSAR Questions

6.6	Fan Cooler Proof Testing
6.14	Fan Cooler Motor Testing
6.15	Fan Cooler Damper Operation
6.16	Spray Pump NPSH
14.2	Hydrogen Generation During Post-LOCA
14.16	Off-gassing from Containment Coatings
14.17	Hydrogen Control Systems

STATEMENT OF GDC 42 - INSPECTION OF CONTAINMENT CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

EVALUATION OF COMPLIANCE

1. FSAR, Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 42. Refer to response to Criterion 39.

2. General Review

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components.

a. Zion Station Radiological Safety Technical Specifications, Section 4.5

The surveillance requirements for the containment fan coolers are given in Section 4.5 of the Zion Radiological Safety Technical Specifications.

b. Zion Station Radiological Safety Technical Specifications, Section 4.6

The surveillance requirements for the containment spray system are given in Section 4.6 of the Zion Radiological Safety Technical Specifications.

c. Zion Station Radiological Safety Technical Specifications, Section 4.8.8

The surveillance requirements for the hydrogen control system are given in Section 4.8.8 of the Zion Radiological Safety Technical Specifications.

3. Conclusion

The Zion Station design addresses the criteria contained in GDC 42.

4. References

FSAR Sections

6.3	Containment Fan Cooler System
6.4	Containment Spray System
6.8	Recombiner
14.3.6	Controlled Containment Venting after a LOCA

FSAR Questions

6.10 Inservice Inspection

Zion Radiological Safety Technical Specifications

4.5 Reactor Containment Fan Coolers Surveillance Requirements
4.6 Containment Spray Surveillance Requirement
4.8.8 Hydrogen Control System Surveillance Requirement

STATEMENT OF GDC 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 43. Refer to response to Criterion 37.

2. General Review

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing as detailed above.

a. Zion Station Radiological Safety Technical Specifications, Section 4.5

The testing requirements for the reactor containment fan coolers are given in Section 4.5 of the Zion Radiological Safety Technical Specifications.

b. Zion Station Radiological Safety Technical Specifications, Section 4.6

The testing requirements for the reactor containment spray system are given in Section 4.6 of the Zion Radiological Safety Technical Specifications.

c. Zion Station Radiological Safety Technical Specifications, Section 4.8.8

The testing requirements of the hydrogen control system are given in Section 4.8.8 of the Zion Radiological Safety Technical Specifications.

3. Conclusion

The Zion Station design addresses the criteria contained in GDC 43.

4. References

FSAR Sections

6.3	Containment Fan Cooler System
6.4	Containment Spray System
6.8	Recombiner
14.3.6	Controlled Containment Venting After a LOCA

FSAR Questions

6.10	Inservice Inspection

Zion Station Radiological Safety Technical Specifications

4.5	Reactor Containment Fan Coolers Surveillance Requirements
4.6	Containment Spray Surveillance Requirements
4.8.8	Hydrogen Control System Surveillance Requirements

STATEMENT OF GDC 44 - COOLING WATER

A system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 44. The component cooling and service water systems provide appropriate cooling capacity for structures, systems, and components important to safety and are designed with appropriate redundancy. A single failure can be accommodated without impairing the safety function of the systems. Appropriate leak detection capability is provided.

2. General Review

A system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink shall be provided.

a. FSAR, page 9.3-1

The component cooling system is designed to remove heat from the residual, spent fuel pit, seal water, letdown, excess letdown and sample heat exchangers, the residual heat removal and reactor coolant pumps, and waste disposal systems. Component cooling water is cooled by the service water system.

b. FSAR, page 9.6-1

The service water system supplies all the equipment cooling water for the plant, including the emergency shutdown requirements.

The system function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

a. FSAR, page 9.3-2

During normal operation, two out of the five component cooling pumps and two out of the three heat exchangers are capable of serving all operating components in both units. Three pumps and two heat exchangers are required for the removal of residual and sensible heat during cooldown. In the event of a LOCA in one plant, one pump and one heat exchanger are capable of fulfilling system requirements.

b. FSAR, page 9.6-2

Normal operation requires two service water pumps on each unit with the third pump serving as a standby. Under emergency shutdown and accident conditions one pump is required for each unit.

Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished assuming a single failure.

a. FSAR, page 9.3-1, Table 8.4-2, and Figure 8.4-1

Active component cooling system components considered vital to the cooling function are redundant. Any active or passive failure in the system will not prevent the system from performing its design function. Component cooling water pumps are on the ESF Bus. The system design provides means for detection of radioactivity entering the system from the reactor coolant system and its associated auxiliary systems, and includes provision for isolation of system components.

b. FSAR, page 9.6-1

The six service water system pumps feed two separate main supply headers. The headers are crosstied so that any combination of pumps can serve both units under normal operating conditions.

c. FSAR, page 9.6-3

The power supply for the service water system is arranged so that one pump is on each of six separate essential buses. Each of the five diesel generator units is sized to accommodate one service water pump in addition to the other vital engineered safeguard loads to be used in the event of a loss of normal auxiliary power.

d. FSAR, page 9.6-4

Those portions of the service water system that supply cooling water to critical systems have been designed to provide isolation capability under postulated leakage conditions.

3. Conclusions

The Zion design addresses the criteria contained in Criterion 44.

4. References

FSAR Sections

9.3	Component Cooling System
9.6	Service Water System

FSAR Questions

9.1	Cooling Water Availability

STATEMENT OF GDC 45 - INSPECTION OF COOLING WATER SYSTEMS

The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

There are no specific design provisions for inspecting the component cooling water or service water systems other than those required for routine maintenance. Both systems are in continuous operation. This fact, coupled with the leak detection provisions employed, makes such design provisions unnecessary at Zion.

2. General Review

The cooling water system shall be designed to permit appropriate periodic inspection of important components to assure integrity and capability of the system.

a. FSAR, page 9.3-9

Active components of the component cooling system are either in continuous or intermittent use during normal plant operation. Periodic visual inspections and preventative maintenance are conducted using normal industrial practice.

b. Zion Station Radiological Safety Technical Specifications, Section 4.8.6

The surveillance requirements for the component cooling system are given in Section 4.8.6 of the Zion Radiological Safety Technical Specifications.

c. Zion Station Radiological Safety Technical Specifications, Section 4.8.7

The surveillance requirements for the service water system are given in Section 4.8.7 of the Zion Radiological Safety Technical Specifications.

3. Conclusion

The Zion Station design conforms with the intent of Criterion 45.

4. References

FSAR Sections

9.3 Component Cooling System
9.6 Service Water System

FSAR Questions

6.10 Inservice Inspection

Zion Radiological Safety Technical Specifications

4.8.6 Component Cooling System Surveillance Requirements
4.8.7 Service Water System Surveillance Requirements

STATEMENT OF GDC 46 - TESTING OF COOLING WATER SYSTEMS

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 46. As stated in response to Criterion 45, both service and component cooling water systems are in continuous operation. Periodic testing to supplement this operation is provided for in the technical specifications. However, operational testing of the system will be performed in one phase while the tests of actuation and power transfer will occur in a separate phase.

2. General Review

   a. Zion Station Radiological Safety Technical Specifications, Section 4.8.6

      The testing requirements for the component cooling system are given in Section 4.8.6 of the Zion Radiological Safety Technical Specifications.

   b. Zion Station Radiological Safety Technical Specifications, Section 4.8.7

      The testing requirements for the service water system are given in Section 4.8.7 of the Zion Radiological Safety Technical Specifications.

3. Conclusions

The Zion Station design conforms with the intent of Criterion 46.

4. References

   FSAR Sections

   9.3 Component Cooling System
   9.6 Service Water System

Zion Station Radiological Safety Technical Specifications

REACTOR CONTAINMENT

STATEMENT OF GDC 50 - CONTAINMENT DESIGN BASIS

The reactor containment structure, including access openings, penetrations, and the containment heat removal system, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect considerations of (1) the effects of potential energy sources that have not been included in the determination of the peak conditions, such as energy in steam generators and, as required by 10 CFR 50.44, energy from metal-water and other chemical reactions thay may result from degradation, but not total failure, of emergency core cooling functioning; (2) the limited experience and experimental data available for defining accident phenomena and containment responses; and (3) the conservatism of the calculational model and input parameters.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 50. The containment structure has been designed with sufficient margins.

2. General Review

The reactor containment structure, including access openings, penetrations, and the containment heat removal system, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

a. FSAR, page 5.1-4

The design pressure (47 psig) and temperature (271° F) of the containment are in excess of the peak pressure (42 psig) and temperature (263° F) occuring as the result of the complete blowdown of the reactor coolant system up to and including the hypothetical, double-ended severance of a reactor coolant pipe.

The containment structure and all penetrations are designed to withstand the loadings of the design basis accident with the combined design or maximum potential seismic condition.

b. FSAR, page 5.1-34

The containment vessel was designed for the following specific loading cases:

1. D + F + L,

2. D + F + L + W (or E),

3. D + F + L + P + T + W (or E), and

4. D + F + L + p'

Where:

D = dead load,

L = appropriate live load,

F = appropriate prestressing load (varies with time between initial and final prestress loads),

P = design pressure,

P' = test pressure,

W = wind load,

E = design seismic load, and

T = thermal loads based on a temperature corresponding to pressure P.

c. FSAR, page 14.3.5-8

The containment is designed to leak at a rate of less than 0.1 percent per day at design pressure without including the benefit of either the isolation valve seal water system or the penetration pressurization system.

This margin shall reflect consideration of (1) the effect of potential energy sources that have not been included in the determination of peak conditions..., (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservativeness of the calculational model and input parameters.

a. FSAR, page 5.1-4

The design pressure is not exceeded during any subsequent long-term pressure transient as determined by the combined effects of heat sources such as residual heat and metal-water reactions with minimum operation of the emergency core cooling and the containment fan cooler system.

b. FSAR, page 14.3.4-2

All the intial core stored energy and the power generated by the core during blowdown is available for transfer to the coolant, and thence to the containment. The intitial metal sensible energy is transferred to the coolant by a time-dependent temperature difference calculation. It should be emphasized that the energy transferred from the core for the containment evaluation far exceeds that transferred from the core thermal evaluation. That is to say a conservatively high core heat transfer coefficient is used for the containment evaluation, while a conservatively low coefficient is used during core thermal evaluation.

3. Conclusion

The Zion Station design addresses the criteria contained in Criterion 50.

4. References

FSAR Sections

5.1 Containment System Structure
14.3.4 Containment Integrity Evaluation
14.3.5 Environmental Consequences of a Loss of Coolant Accident

FSAR Questions

5.1 through 5.75

STATEMENT OF GDC 51 - FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion design conforms with the intent of Criterion 51. The containment ferritic materials are selected to insure that their temperature under normal operating and testing conditions will be at least 30° F above the NDTT. Detailed stress analyses have been made of the containment liner and liner anchors under normal and postulated accident conditions. Code allowable material discontinuities have been considered.

2. General Review

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of a rapidly propagating fracture is minimized.

a. FSAR, page 5.1-18

Dynamic test results of representative tendon assemblies are made based on 500 cycles of rapid loading from stress level 0.70 of the tensile strength of the prestressing steel (f') to stress level 0.75 f' and return to 0.70 f'. Earthquake, wind, and design basis accident condition loadings will not generate more than 100 cycles of maximum stress variations during the life of the plant.

b. FSAR, page 5.1-23

Each of the tendons has been pretested at the time of initial jacking, and the stress in the tendons under accident loading is only 80% of this jacking stress. This means that a failure of a tendon under design accident loading is quite remote.

c. FSAR, page 5.1-26

The liner plate is designed as a leaktight membrane and is not relied upon to assist the concrete containment in maintaining structural integrity. The liner is constructed on 1/4 inch A442 Grade 60 carbon steel. This material is readily weldable and has good ductility which insures its capacity for leaktightness under all expected strains.

d. FSAR, page 5.1-39

In consideration of the load factors presented in Subsection 5.1.2.4, the load combinations are less than the yield strength of the structure which is defined as the upper limit of elastic behavior of the effective load carrying materials.

The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties; (2) residual, steady state, and transient stresses; and (3) size of flaws.

The strength of the containment structure of working stress and overall yielding was equated to various combinations of loadings to insure its integrity. Proper performance of the containment structure was examined with respect to strength, the nature and the amount of cracking, the magnitude of deformation, as well as corrosion. The containment structure was designed and analyzed to meet performance and strength requirements under the conditions listed in Subsection 5.1.2.4 (FSAR, page 5.1-32).

3. Conclusions

The Zion Station design addresses the criteria contained in Criterion 51.

4. References

FSAR Sections

5.1 Containment System Structure

FSAR Questions

5.17 Safety of Containment Structure ACI-318-63
5.18 ACI-318-63
5.19 Yield Strength
5.20 Stress Analysis
5.21 Checking Design
5.22 Prestressing Tendon Anchors
5.23 Elastic Low Strain Behavior

STATEMENT OF GDC 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 20, May 1972

The Zion Station design conforms with the intent of Criterion 52. The Zion Station design however, incorporates a continuous pressurization system for weld channels and penetrations that permits on-line surveillance of containment integrity. This system offers a significant improvement over periodic pressure testing in that information is continuously available regarding containment integrity and no periodic testing of the entire containment is necessary. In addition, the provisions of 10 CFR 50, Appendix J, are met at Zion regarding required integrated testing.

2. General Review

Integrated leak rate tests are performed in accordance with 10 CFR 50, Appendix J, except where specific exemptions in accordance with 10 CFR 50.12 have been requested.

Integrated leak rate tests are conducted at 47 psig, the containment design pressure (FSAR, page 5.1-4).

3. Conclusion

Zion Station design addresses the criteria of GDC 52.

4. References

a. Radiological Safety Technical Specifications for Zion Station Units 1 & 2, Sections 3.10.1 and 4.10.1, Containment Leakage Rate Testing.

b. FSAR Section 5.1.1.1 Principal Design Criteria.

c. G. J. Pliml, CECo, letter to K. R. Goller, NRC, September 26, 1965.

d. A. Schwencer, NRC, letter to R. L. Bolger, CECo, November 23, 1976.

e. D. E. O'Brien, CECo, letter to A. Schwencer, NRC, December 17, 1976.

f. D. E. O'Brien, CECo, letter to K. R. Goller, NRC, January 31, 1977.

g. A. Schwencer, NRC, letter to R. L. Bolger, CECo, January 26, 1977.

h. D. E. O'Brien, CECo, letter to A. Schwencer, NRC, February 16, 1977.

i. D. E. O'Brien, CECo, letter to A. Schwencer, NRC, August 31, 1977.

j. A. Schwencer, NRC, letter to R. L. Bolger, CECo, September 17, 1977.

k. W. F. Naughton, CECo, letter to H. R. Denton, NRC, July 7, 1980.

STATEMENT OF GDC 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10 Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 53. Online surveillance of containment integrity is a feature of the Zion design. In addition, access for routine maintenance is possible for those areas of particular importance, such as penetrations.

2. General Review

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing of containment design pressure of the leaktightness of penetrations that have resilient seals and expansion bellows.

a. FSAR, page 5.2-17

Provisions are made for surveillance of the plant after startup.

Connections for compressed air and for pressure gauges were capped after the preoperational containment pressure test to permit repetition of the test at some future time.

b. Zion Station Radiological Safety Technical Specifications, Section 4.10

In Section 4.10 of the Zion Radiological Safety Technical Specifications, the containment surveillance program is discussed.

3. Conclusion

The Zion Station design addresses the criteria contained in Criterion 53.

4. References

FSAR Sections

5.2.3 Surveillance

FSAR Questions

5.53 Access Provisions
5.56 Inspection Program

Zion Radiological Safety Technical Specifications

4.10 Containment Structural Integrity Surveillance Requirements

STATEMENT OF GDC 54 - PIPING SYSTEMS PENETRATING CONTAINMENT

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within accepted limits.

EVALUATION OF COMPLIANCE

1. FSAR, Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 54. Appropriate isolation capability is provided for piping systems penetrating the containment. In addition, the Zion Station design includes an isolation valve seal water system as further assurance of containment integrity.

2. General Review

Piping systems penetrating primary containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities that reflect the importance to safety of isolating these piping systems.

a. FSAR, page 6.6-1

All lines for which isolation is required are provided with two barriers so that no single failure will prevent operation.

b. FSAR, pages 6.6-2 through 6.6-5

Penetrations have been divided into seven classes, discussed on Zion FSAR pages 6.6-2 through 6.6-5.

c. FSAR, page 6.5-1

Supplemental leakage detection provisions, when necessary, are sufficiently sensitive so that any increase in leakage rates can be detected while the total leakage rate is still below a value consistent with safe operation of the plant.

Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

a. FSAR, page 6.6-6

All automatic containment isolation valves can be manually closed from the control room.

Pneumatic leak tests can be periodically performed on all containment isolation valves with the exception of those valves which, under postaccident containment isolation conditions, are expected to be maintained continually at a pressure equal to or greater than the containment peak accident pressure.

b. Zion Station Radiological Safety Technical Specifications, Section 4.9

The testing program for the containment isolation valves is discussed in Section 4.9 of the Zion Radiological Safety Technical Specifications.

3. Conclusions

The Zion Station design addresses the criteria contained in Criterion 54.

4. References

FSAR Sections

6.5 Leakage Detection Provisions
6.6 Containment Isolation

FSAR Questions

6.18 Passive Failure in ECCS

Zion Radiological Safety Technical Specifications

4.9 Containment Isolation Systems Surveillance Requirements

STATEMENT OF GDC 55 - REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 55. T'. only lines comprising a portion of the reactor coolant pressure boundary are the reactor coolant sample lines. These lines are equipped with one normally closed valve inside and 2 normally closed isolation valves outside the containment. In veiw of the intermittant use of these lines and their small size (3/8 in.), these isolation provisions are appropriate.

2. General Review

Each line that is part of the reactor coolant pressure boundary and that penetrates containment shall be provided with containment isolation valves as described above.

Each line is equipped with the following: a manual isolation valve close to the sources; an air-operated valve immediately downstream of the isolation valves; containment boundary isolation/trip valves located outside the containment, and manual valves located inside the sampling room (FSAR, page 9.8-2).

3. Conclusions

The Zion Station design addresses the criteria contained in Criterion 55.

4. References

FSAR Sections

6.6 Containment Isolation
9.8 Sampling Systems

STATEMENT OF GDC 56 - PRIMARY CONTAINMENT ISOLATION

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 56, except for the containment pressure and vacuum relief lines. These lines are provided with two automatic isolation valves outside the containment. However, these lines are equipped with isolation valve seal water and are located as close as practicable to the containment. These lines, as do all lines penetrating the containment, fall into a specific class of lines as discussed in Section 6.6 of the FSAR.

2. General Review

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation vaves as described above.

a. FSAR, pages 6.6-2 through 6.6-5

Isolation lines which penetrate containment and connect directly to the containment atmosphere are divided into classes with different isolation methods.

b. FSAR, Table 6.6.5-1

The isolation provided for specific lines is indicated in Zion FSAR Table 6.6.5-1.

Isolation valves outside containment shall be located as close to the containment as practical, and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

a. FSAR, page 6.6-2

Isolated lines between the containment and the first outside isolation valve are designed to the same seismic criteria as the containment vessel and are considered as an extension of the containment.

b. FSAR, page 6.6-6

Air-operated valves fail closed in the event of loss of air or loss of power.

3. Conclusions

The Zion Station design conforms with the intent of Criterion 56, except as discussed in Section 1 of Evaluation of Compliance.

4. References

FSAR Sections

6.6 Containment Isolation
9.10.6 Normal Containment Ventilation System

STATEMENT OF GDC 57 - CLOSED SYSTEM ISOLATION VALVES

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 57. At least one isolation valve, not a check valve, is provided on all closed system lines penetrating the containment.

2. General Review

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

a. FSAR, page 6.6-3

Incoming and outgoing lines which penetrate the containment and are connected to closed systems are provided with at least one manual isolation valve outside containment.

b. FSAR, page 6.6-4

Lines that penetrate the containment and can be opened to containment but are normally closed during reactor operation are provided with two isolation valves in series, either inside or outside the containment.

c. FSAR, page 6.6-2

Isolation lines between the containment and the first outside isolation valve are designed to the same seismic criteria as the containment vessel, and are considered as an extension of the containment.

d. FSAR, Table 6.6.5-1

The isolation provided for specific lines is indicated in FSAR Table 6.6.5-1.

3. Conclusions

The Zion Station design conforms with the intent of Criterion 57.

4. References

FSAR Sections

6.6 Containment Isolation

FUELS AND RADIOACTIVITY CONTROL

STATEMENT OF GDC 60 - CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, Dec., 1971

The Zion Station design conforms with the intent of Criterion 60. An extensive treatment system has been incorporated in the Zion design for liquid wastes. Gaseous wastes are processed by appropriate holdup systems. Solid wastes are solidified in concrete (except for clothing, paper, etc.) and processed in 55-gallon drums for eventual disposition in licensed burial grounds.

2. General Review

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences.

a. Appendix I Report, Amendment 1, page 1.1-3

The radiological impact of radionuclides released in liquid effuents has been calculated. Consumption factors used are those given in the HERMES Computer Code and summarized in Appendix I Report, Table 1.1-2.

Specific doses are given in Appendix I Report, Table 1.1-7, for the various points of interest listed in Appendix I Report, Table 1.1-6.

All are within Appendix I to 10 CFR 50 guidelines.

b. Appendix I Report, page 1.1-2

The radiological impact of radionuclides released in gaseous effluents has been calculated. Consumption factors used are those given in the HERMES computer code and summarized in Appendix I Report, Table 1.1-2. Specific doses at various points of interest are given in Appendix I Report, Tables 1.1-3 and 1.1-4. All calculated doses are within Appendix I to 10 CFR 50 guidelines.

c. Letter from R. L. Bolger to A. Schwencer, dated April 26, 1977

A discrepancy in the noble gas source item was identified in Reference b. This was investigated and results reportd to the NRC in Reference c. The releases were well within Appendix I deadlines.

Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

a. Appendix I Report, Table 1.1-2

Consumption factors for maximum exposed individuals are listed in Appendix I Report, Table 1.1-2. These consumption factors were taken from (1) Regulatory Guide 1.109, Table A-2, (2) calculated from the HERMES computer code, and (3) from HERMES are used in Commonwealth Edison's annual and semiannual reports on station radioactive waste environmental monitoring and occupational personnel radiation exposure.

b. Appendix I Report, Items 2.3 and 2.6

The values of X/Q and D/Q for the various points of interest are given in Item 2.3 of the Appendix I Report. The calculations make use of the joint frequency distribution data as discussed in Item 2.6 of the Appendix I Report.

c. Appendix I Report, page 1.1-6

Radiation doses to man from radionuclides in liquid effluents may result from many pathways. For this station, the pathways and aquatic dispersion factors considered are shown in Table 1.1-6.

d. Appendix I Report, Figure 1.1-1 and Item 2.1

Schematically, the station systems affecting gaseous effluents are shown in Figure 1.1-1 of the Appendix I Report. More detailed P&ID's and operating parameters are given in Item 2.1 of the Appendix I Report.

e. Appendix I Report, page 1.1-3

Liquid effluents are released from the plant into a discharge stream which has an average flow of 21×10^5 gpm. This stream discharges into Lake Michigan.

3. Conclusion

All aspects of Criterion 60 were addressed in the Zion Station design and approved by the NRC in the OL review of the plant.

4. References

a. "Information Relevant to Keeping Levels of Radioactivity in Effluents to Unrestricted Areas As Low As Reasonably Achievable," Zion Station Units 1&2, June 4, 1976 and Amendment 1, November 12, 1976 (Appendix I Report).

b. A. Schwencer, NRC, letter to R. L. Bolger, CECo, January 26, 1977.

c. R. L. Bolger, CECo, letter to A. Schwencer, NRC, April 26, 1977.

STATEMENT OF GDC 61 - FUEL STORAGE AND HANDLING AND RADIO-ACTIVITY CONTROL

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December 1971

The Zion Station design conforms with the intent of Criterion 61, except that inspection and testing are limited to those associated with routine maintenance. Surveillance of safety-related items is accomplished by virtue of routine monitoring of the day to day operations of the systems.

Appropriate shielding, filtration, heat removal, and water inventory control provisions have been included for these systems.

2. General Review

Capability to Permit Periodic Inspection and Testing of Components

a. FSAR, page 9.5-6

The active components of the system are in continuous use during normal plant operation and no additional periodic tests are specified. Periodic visual inspections and preventive maintenance are conducted following normal industrial practice.

b. FSAR, page 9.7-17

Prior to initial fueling, preoperational checkouts of the fuel handling equipment are performed to ensure proper performance of the fuel handling equipment and to familiarize plant operating personnel with operation of the equipment.

Prior to subsequent refueling operations, the equipment is inspected for operating condition, and certain components, such as the fuel transfer car and manipulator crane, are operated to ensure reliable performance prior to moving irradiated fuel.

Shielding for Radiation Protection

c. FSAR, page 11.2-4

At the fuel assembly tilting device, a spent fuel assembly is shielded in the direction of the primary loop compartment by 4 ft-4 in. of water and 4 ft of concrete which reduces the dose rate at 4 days decay to less than 10 mrem/hr in this compartment. At other locations and in other directions, the shielding is greater.

d. FSAR, page A-30

Shielding is provided for fuel handling and waste storage areas to lower radiation doses to levels below limits specified in 10 CFR 20.

e. FSAR, page 9.7-3

Adequate shielding for radiation is provided during reactor refueling by conducting all spent fuel transfer and storage operations under water. This permits visual control of the operation at all times while maintaining low radiation levels, less than 2.5 mrem/hr, for periodic occupancy of the area by operating personnel.

Containment, Confinement, and Filtering Systems

f. FSAR, page 9.7-4

The reactor cavity, refueling canal, and spent fuel storage pit are reinforced concrete structures with seam-welded stainless steel plate liners. These Class I structures are designed to withstand the anticipated earthquake loadings and to prevent liner leakage even in the event the reinforced concrete develops cracks.

g. FSAR, pages 9.7-9 and 9.7-10

The refueling cavity is a reinforced concrete structure that forms a pool above the reactor when it is filled with borated water for the refueling.

The spent fuel storage pit is constructed of reinforced concrete. The entire interior basin face and transfer canal are lined with stainless steel.

A separate area with wing walls is provided at the end of the pool for the storage of the spent fuel cask. The base mat of the fuel storage pool in the cask storage area has been designed so that structural integrity will not be lost in the event of a cask drop.

h. FSAR, page 9.10-2

The supply system filters 100% outdoor air in two stages where the final stage has a nominal efficiency of 85% based upon the NBS atmosphere dust pool test. The filtered air is heated or cooled as required to maintain a nominal supply air temperature of 75° ± 10° F.

All exhaust air which is returned from t auxiliary building and fuel handling building is filtered through HEPA filters which are tested on site for a nominal bank efficiency of 99.95% based on 0.3-micron DOP tests.

i. FSAR, page 9.10-8

The fuel handling building exhaust filters are composed of banks of prefilters and HEPA filters installed in series. Each filter unit has a rated flow of 22,000 cfm. Each prefilter bank contains 24 individual filter elements rated at 85% efficiency of 99.7% based on the NBS atmospheric dust spot test. Each HEPA filter banks contains 24 individual filter elements each having a nominal efficiency of 99.7% based on the DOP test.

Residual Heat Removal

j. FSAR, page 9.5-1

The spent fuel pit cooling system has two cooling trains.

k. Licensing Report, Zion Nuclear Power Plant, Units 1 & 2, "Spent Fuel Rack Modification", February 3, 1978 Section 3.6

Spent fuel pool cooling capability for the new spent fuel pool modification increasing the number of storage spaces to 2112 is discussed in Section 3.6 of the licensing report for the Zion Nuclear Power Plant, "Spent Fuel Rack Modification." This analysis showed a maxi um change in temperature calculated for the peak pow r fuel assembly was 32.38° F based on a bulk pool temperature of 120° F.

l. FSAR, page 9.7-2

The refueling water provides reliable and adequate cooling medium for spent fuel transfer.

m. FSAR, page 9.7-4

The reactor is refueled with equipment designed to handle the spent fuel under water from the time it leaves the reactor until it is placed in a cask for shipment from the site. Underwater transfer of spent fuel provides an effective, economic, and transparent radiation shield, as well as a reliable cooling medium for removal of decay heat. Boric acid is added to the water to ensure subcritical conditions during refueling.

Prevention of Fuel Storage Coolant Reduction

System piping is arranged so that failure of any pipeline does not drain the spent fuel pit below the top of the stored fuel elements (FSAR, page 9.5-1).

The most serious failure of this system would be complete loss of water in the storage pit. To protect against this possibility, the spent fuel pit cooling connections enter near the water level so that the pit cannot be gravity-drained (FSAR, page 9.5-1).

3. Conclusions

The Zion Station design addresses the criteria contained in GDC 61.

4. References

a. FSAR Sections

9.5 Spent Fuel Pit Cooling

9.7 Reactor Components and Fuel Handling System

9.10 Plant Ventilation

Reports

Licensing Report, Zion Nuclear Power Plant Units 1 & 2, "Spent Fuel Rack Modification," February 3, 1978.

Letters

A. Schwencer, NRC, letter to D. L. Peoples, CECo, February 28, 1980, Containing License Amendment Nos. 52 and 49 for Zion Units 1 and 2.

STATEMENT OF GDC 62 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December, 1971

The Zion Station design conforms with the intent of Criterion 62. Fuel storage and transfer systems are configured so as to preclude criticality.

2. General Review

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

a. FSAR, page A-29

Criticality in new and spent fuel storage areas is prevented both by physical separation of new and spent fuel elements and the presence of borated water in the spent fuel storage pit.

b. Zion Station Technical Specifications, Section 5.5.2, page 299

Irradiated fuel assemblies will be stored prior to offsite shipment in the stainless steel-lined fuel pool which is located in the fuel handling building. Borated water is used to fill the spent fuel storage pit at a concentration to match that used in the reactor cavity and refueling canal during refueling operations. The fuel is stored in a vertical array with a nominal center spacing of 10.35 inches between assemblies to assure a K effective of less than 0.95 even if unborated water is used to fill the pit, for fuel having maximum loading of 40.6 gms. of U-235 per axial centimeter of fuel assembly length (about 3.2 weight percent U-235).

3. Conclusion

The Zion Station design adresses the criteria contained in Criterion 62.

4. References

FSAR Sections

9.7 Reactor Components and Fuel Handling System

Zion Radiological Safety Technical Specifications

5.5.2 Spent Fuel Storage

Reports

Licensing Report, Zion Nuclear Power Plant Units 1 & 2 "Spent Fuel Rack Modification", February 3, 1978.

Letters

A. Schwencer, NRC, letter to D. L. Peoples, CECo, February 28, 1980, Containing License Amendment Nos. 52 and 49 for Zion Units 1 and 2.

STATEMENT OF GDC 63 - MONITORING FUEL AND WASTE STORAGE

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December, 1971

The Zion Station design conforms with the intent of Criterion 63. Fuel storage areas and waste storage areas are provided with radiation monitors which alarm at abnormally high levels. Appropriate action may be initiated upon such alarms. The spent fuel pit cooling system is furnished with a temperature alarm and other system instrumentation to indicate off-normal operation.

2. General Review

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

a. FSAR, page 9.7-3

An area radiation monitor of the gamma scintillation type is provided to monitor the spent fuel pool area. Temperature and level instruments are provided to guard against the loss of cooling capability.

b. FSAR, page 9.7-16

Gamma radiation levels in the containment and fuel storage areas are continuously monitored. These monitors provide an audible alarm at the initiating detector indicating an unsafe condition.

c. FSAR, page 11.3-6

A scintillation counter inserted in an in-line well is used as the detector. The output is preamplified and transmitted to the radiation monitoring system cabinets in the control rooms. The activity level is indicated on a meter and recorded. High-activity alarm indications are displayed on the control board annunciator in addition to the radiation monitoring cabinets.

A high activity alarm indicates a leak in any heat exchanger tube in the spent fuel pit cooling and volume control system and residual heat removal system or a cooling coil for the thermal barrier cooler on a reactor coolant pump.

d. Radiation Monitoring, FSAR, page Q11.1-3

1. Type 1 - applicable channel: fuel handling building (pool area). Upon a high radiation trip, ventilation air is transferred through charcoal filters for iodine removal before being released to the stack. The transfer is effected by automatically starting the pertinent fans and switching corresponding dampers to the proper position.

2. Type 2 - the fuel handling building (pool area) monitor is interlocked with the spent fuel pool crane in such a manner as to prevent lifting of spent fuel elements from the pool.

3. Conclusion

All aspects of Criterion 63 were addressed in the Zion Station design and approved by the NRC in the OL review of the plants.

4. References

FSAR Sections

9.7 Reactor Components and Fuel Handling System

11.3 Radiation Monitoring System

FSAR Questions

11.1 Radiation Monitoring

11.3 Area Radiation Monitors

STATEMENT OF GDC 64 - MONITORING RADIOACTIVITY RELEASES

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

EVALUATION OF COMPLIANCE

1. FSAR Question 1.10, Amendment 18, December, 1971

The Zion Station design conforms with the intent of Criterion 64. Extensive monitoring systems, both in the plant and in the environs, have been furnished at Zion.

2. General Review

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

a. FSAR, page 11.3-1

The radiation monitoring system is designed to continuously monitor the containment atmosphere, all plant effluents, and various in-plant locations to provide plant operators with indication and warning in areas where radioactive sources exist and operating personnel are normally required to be present. In addition, this system will also serve to give an immediate alarm in case of any significant change in the level of radioactivity in the containment, the spent fuel storage area, or the auxiliary building.

b. FSAR page A-8

The facility contains means for monitoring the containment atmosphere, effluent discharge paths, and the facility environs for radioactivity which could be released under any conditions.

c. FSAR page 2.8-1

The environs monitoring during operation of the Zion Station will follow a pattern similar to the preoperational program.

d. <u>FSAR, page 2.8-2</u>

The operational environmental survey will incorporate measurements to provide background data and measure possible plant effects.

The present radiological monitoring program at Zion Station was designed in 1969 and sample collection was started in March, 1970. Data and experience from the program at Zion, Quad Cities, and Dresden Stations and analysis of these programs have indicated that the following changes to the Zion radiological monitoring program can be made.

1. <u>Gamma Background - Ion Chambers</u>

   No changes.

2. <u>Filmbadges</u>

   Filmbadges have been dropped from the program because the badges are insensitive to ambient background and expected operational radiation levels and are subject to erroneous readings due to changing weather conditions. Filmbadges were used to measure external gamma radiation, but their function is more than adequately covered by ion chambers and thermoluminescent dosimeters (TLD).

3. <u>TLD</u>

   The sampling schedules for TLD's have been increased from one set semiannually and one set annually to one set quarterly and one set annually.

4. <u>Airborne Particulates</u>

   The frequency of airborne particulate sampling and gross alpha radioactivity analysis has been reduced from weekly to monthly because airborne alpha is not expected to be a constituent of station emissions.

5. <u>Airborne Iodine-131</u>

   The frequency of airborne iodine sampling has been reduced from weekly to every two weeks to increase the sensitivity of detection. If A_1 is the activity collected at the end of the first week of sampling, then $A_2 + A_1 e^{-\lambda t} = A_2 + A_1/2 = 3A_1/2$ is the activity (approximate) collected at the end of two weeks, assuming that the release of iodine is continuous. At the end of this period it is easier to detect the latter activity $3A_1/2$ than it is to detect the activity A_1.

6. Private Well Water

The frequency of sampling of private well water has been reduced from monthly to quarterly because of the improbability of interaction of Zion Station liquid effluents released to Lake Michigan and well water. The ground water flows toward the lake.

7. Public and Surface Waters

Composites of weekly collected public and surface water samples collected weekly will be analyzed quarterly rather than monthly for tritium because tritium concentrations in the waters near Zion are fairly constant and the long halflife of the tritium precludes a loss of significant amounts of activity over a 3-month period. Thus, nothing is gained by collecting samples more frequently than on a quarterly basis. The frequency of Sr-90 analyses of composited surface water samples will be increased to semiannually from its former schedule of "special" analyses.

8. Sediment

Sediment is sampled as an "indicator" medium because it is believed to be a reservoir from the accumulation of the longer life radionuclides. For these nuclides, such as Cs-137 or Sr-90, the change in frequency of sampling from quarterly to semiannually will not affect the ability to measure significant accumulations, if any.

9. Benthic Organisms

The frequency of sampling of benthic organisms has been reduced from quarterly to semi-annually because of the difficulty of finding sufficient quantities for radioactivity analysis on a quarterly basis. In the winter, collection is prevented because of ice cover, and in the rest of the year, benthic populations are extremely sparse in the Zion Station area. Collections large enough for analysis can be made, if at all, only during the primary growth periods which occur twice a year, in the spring and fall.

Alpha radioactivity will not be routinely measured in benthic organisms, because alpha-emitting radionuclides have not been detected as constituents of plant liquid wastes and are not concentrated in the environment. Therefore, they would not be detectable in the benthos.

10. Periphyton

The frequency of sampling of periphyton has been reduced from quarterly to thrice annually because ice covers Lake Michigan and local rivers during the winter.

Alpha radioactivity will not be routinely measured in periphyton, for the same reason that it was discontinued for benthos.

11. Fish

The frequency of fish sampling and analysis has been increased to thrice annually from semi-annually. The analysis has been augemented by dividing each fish sample by species into two portions ...whole and filet for gamma emitters and the routine measurement of Sr-89 and Sr-90. The whole sample analysis will measure the maximum possible radioactivity uptake and the added filet analysis will indicate typical radioactivity uptakes by humans.

Alpha analyses will not be conducted on these samples, for the same reason that it was discontinued for benthos.

12. Milk

The milk sampling program has been increased to weekly sampling from April to September rather than the previously scheduled monthly sampling.

13. Grass

Because grass does not grow and is not used to graze cattle in the cold winter months, the period of sampling grass has been limited to the time period when grass is available, April to September.

14. Cattle Feed and Hay

No changes.

15. Soil

Alpha analyses will not be conducted on soil samples because alpha emitters have not been found in airborne emissions from nuclear stations, and there is no possibility of concentration.

16. Vegetables

The vegetable sampling program will be increased by the addition of routine analyses for Sr-89 and Sr-90; alpha analyses will not be conducted for the same reason that it was discontinued for soil.

17. Precipitation and Dry Deposited Matter

Analysis for grass alpha radioactivity will not be conducted. Alpha radioactivity is not present in airborne emissions.

e. FSAR, Question 11.22

The methods used in quantifying routine effluent releases to the environment consist of continuous radiation monitoring systems and the laboratory analysis of grab samples for each effluent stream.

f. SER, page 11-7

We conclude that the Applicant's environmental monitoring program will be adequate to monitor the radiological impact of plant operation on the environs and to assess the health and safety aspects of the release of radioactivity to the environment from the proposed operation of the plant.

3. Conclusion

All aspects of Criterion 64 were addressed in the Zion Station design and approved by the NRC in the OL review of the plant.

4. References

FSAR Sections

2.8 Environment Radioactivity Studies

11.1 Waste Disposal System

11.3 Radiation Monitoring System

Appendix A General Design Criteria

Zion Station Radiological Technical Specifications

4.9 Containment Isolation Systems Surveillance Requirements

4.10 Containment Structural Integrity Surveillance Requirements

FSAR Questions

2.16 Environmental Monitoring

11.1 Radiation Monitoring

11.4 Radiation Monitoring

11.6 Radiation Monitoring

11.14 Effluent Sampling

11.15 Effluent Sampling

11.20 Radiation Monitoring

11.22 Radiation Monitoring

11.23 Radiation Monitoring

11.38 Radiation Sampling

SER Sections

11.4 Offsite Radiological Monitoring Program

APPENDIX B TO 10 CFR 50 - QUALITY ASSURANCE CRITERIA FOR NUCLEAR POWER PLANTS AND FUEL REPROCESSING PLANTS

The quality assurance program for Zion Station Units 1 & 2 is presented in Commonwealth Edison Topical Report CE-1-A, Revision 13, "Quality Assurance Program for Nuclear Generating Stations." This topical report complies with 10 CFR 50, Appendix B, and has been accepted by the NRC as indicated in the letter from W. P. Haas to D. L. Peoples, dated August 7, 1980, which states in part:

> We have completed our review of Revisions 11, 12, and 13 to the Commonwealth Edison Company (CECo) Topical Report No. CE-1-A,... You indicated in your transmittal letter that the topical report and this revision are intended to be applicable to nuclear units identified by Docket Numbers... 50-295 (Zion Unit 1), 50-304 (Zion Unit 2)...
>
> Based on our evaluation of the proposed changes described in Revisions 11, 12, and 13, we find that your revised topical report continues to meet the criteria of Appendix B to 10 CFR Part 50 and is therefore acceptable..."

APPENDIX E TO 10 CFR 50 - EMERGENCY PLANS FOR PRODUCTION AND UTILIZATION FACILITIES

Compliance with 10 CFR 50, Appendix E, is documented through the Commonwealth Edison Company Generating Station Emergency Plan (GSEP). The GSEP is an emergency plan applicable to all nuclear generating stations operated by Commonwealth Edison Company and considers the consequences of radiological emergencies, as required by 10 CFR 50, Paragraph 50.34 and Appendix E. Additionally, the GSEP addresses the supplemental guidance provided by the NRC in the form of Regulatory Guide 1.101 (Rev. 1, March 1977) and NUREG-0654 (January 1980).

The GSEP has been in effect since Dresden Unit 1 started operation in 1959. The plan was referenced in the Zion PSAR submittal and has been revised periodically to comply with changing requirements. The current GSEP is dated July 1980, Revision 1, and was submitted to the NRC by letter of L. D. Del George to B. K. Grimes, dated July 30, 1980.

APPENDIX G TO 10 CFR 50 - FRACTURE TOUGHNESS REQUIREMENTS

Compliance with 10 CFR 50, Appendix G, is documented in the Zion Station FSAR; the Zion Station Radiological Safety Technical Specifications; WCAP-8724, "ASME III, Appendix G Analysis of the Commonwealth Edison Company Zion Unit 1 Reactor Vessel;" and WCAP-8727, "ASME III, Appendix G Analysis of the Commonwealth Edison Company Zion Unit 2 Reactor Vessel." The sections of Appendix G that contain technical requirements were evaluated for compliance with those requirements.

The Zion Station dockets (Nos. 50-2 and 50.304) contain information on the vessel weld loca. ons and details of material (plate and weld metal) composition and mechanical properties, including T_{NDT} determined from dropweight tests for the plate material. These dockets also include copies of WCAP-8064 (Unit 1) and WCAP-8132 (Unit 2), which address the metal surveillance programs and preirradiation testing results.

SECTION III - FRACTURE TOUGHNESS TESTS

STATEMENT OF SECTION III - PARAGRAPH A

To demonstrate compliance with the minimum fracture toughness requirements of Section IV and V of this appendix, ferritic materials shall be tested in accordance with the ASME Code Section NB-2300, "Fracture toughness requirements for materials." Both unirradiated and irradiated ferritic materials shall be tested for fracture toughness properties by means of the Charpy V-notch test specified by paragraph NB-2321.2 of the ASME Code. In addition, when required by the ASME Code, unirradiated ferritic materials shall be tested by means of the dropweight test specified by paragraph NB-2321.1 of the ASME Code. P. ovision shall be made for supplemental tests in crucial situations such as that described in Section V - Paragraph C.

EVALUATION OF COMPLIANCE

1. FSAR, Question 4.2

Reactor coolant system boundary materials testing complied with ASME Special Material Testing Requirements current at the construction of the Zion Station.

2. SER, page 5-4

The code current at the time of Zion Station construction was not adequate to establish compliance with the proposed AEC "Fracture Toughness Requirements;" however, technical specification limits were established for operation.

3. Zion Station Radiological Safety Technical Specifications, page 90

The fracture toughness properties of the ferritic materials in a reactor coolant pressure boundary are determined in accordance with Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition.

4. D. E. Obrien, CECo, letter to A. Schwencer, September 7, 1977

The results of dropweight tests on plate materials were submitted to the NRC on September 7, 1977.

STATEMENT OF SECTION III - PARAGRAPH B

Charpy V-notch impact tests and dropweight tests shall be conducted in accordance with the following requirements:

1. Location and orientation of impact test specimens shall comply with the requirements of Paragraph NB-2322 of the ASME Code.

2. Materials used to prepare test specimens shall be representative of the actual materials of the finished component as required by the applicable rules of the construction code under which the component is built pursuant to 10 CFR 50.55a, except that ferritic materials intended for the reactor vessel beltline region shall comply with the additional requirements of Section III - Paragraph C of this appendix.

3. Calibration of temperature instruments and Charpy V-notch impact test machines used in impact testing shall comply with the requirements of Paragraph NB-2360 of the ASME Code.

4. Individuals performing fracture toughness tests shall be qualified by training and experience and shall have demonstrated competency to perform the tests in accordance with written procedures of the component manufacturer.

5. Fracture toughness test results shall be recorded and shall include a certification by the licensee or person performing the tests for the licensee that:

   a. the tests have been performed in compliance with the requirements of this appendix,

   b. the test data are correctly reported and identified with the material intended for a pressure-retaining component,

   c. the tests have been conducted using machines and instrumentation with available records of periodic calibration, and

d. records of the quali[illegible]cations of the individual performing the tests are available upon request.

EVALUATION OF COMPLIANCE

1. FSAR, Question 4.2

Reactor coolant system boundary materials testing complied with ASME Special Material Testing Requirements current at the time of construction of the Zion Station.

2. SER, page 5-4

The code current at the time of Zion Station construction was not adequate to establish compliance with the proposed AEC "Fracture Toughness Requirements;" however, technical specification limits were established for operation.

3. D. E. Obrien, CECo, letter to A. Schwencer, September 7, 1977

The reactor vessel belt line region materials are identified, and mechanical properties were presented to the NRC on September 7, 1977.

STATEMENT OF SECTION III - PARAGRAPH C

In additon to the test requirements of Section III - Paragraph A of this appendix, tests on materials of the reactor vessel beltline shall be conducted in accordance with the following minimum requirements:

1. Charpy V-notch (C_V) impact tests shall be conducted at appropriate temperatures over a temperature range sufficient to define the C_V test curves (including the uppershelf levels) in terms of both fracture energy and lateral expansion of specimens. Location and orientation of impact test specimens shall comply with the requirements of Paragraph NB-2322 of the ASME Code.

2. Materials used to prepare test specimens for the reactor vessel beltline region shall be taken directly from excess material and welds in the vessel shell course(s) following completion of the production longitudinal weld joint, and subjected to a heat treatment that produces metallurgical effects equivalent to those produced in the vessel material throughout its fabrication process, in accordance with paragraph NB-2211 of the ASME Code. Where seamless shell forgings are used, or where the same welding process is used for longitudinal and circumferential welds in plates, the test specimens may be taken from a separate weldment provided that such a weldment is prepared using excess material from the shell forging(s) or plates, as applicable, the same heat of filler material, and the same production welding conditions as those used in joining the corresponding shell materials.

EVALUATION OF COMPLIANCE

A Charpy V-notch impact test (Type A) was performed per ASTM E23 which as the current criteria at the time of Zion Station construction. The curve plotted from the test consisted of 15 data points constituting 3 tests at 5 different temperatures (FSAR, page 4A-6).

Materials for Charpy V-notch testing at Zion Station were taken from the core region plates and forgings and core region weldments, including heat-affected zone material (FSAR, page 4A-6).

Commonwealth Edison meets the requirements of Section III - Paragraph C through WCAP-8064 and WCAP-8132, which address the metal surveillance programs and preirradiation testing results for Zion Station Units 1 and 2, respectively. These documents also establish the source, history, and unirradiated test results of metal surveillance specimens. The charpy V-notch test curves cover the temperature range required by Section IV - Paragraph C.1 of Appendix G and are sufficient to define the upper shelf and the changes of ft-lbs and lateral expansion with temperature.

SECTION IV - FRACTURE TOUGHNESS REQUIREMENTS

STATEMENT OF SECTION IV - PARAGRAPH A.1

The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials shall meet the following requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences:

1. The materials shall meet the acceptance standards of paragraph NB-2330 of the ASME Code, and the requirements of Section IV.A.2, 3, 4, and IV.B of this appendix.

EVALUATION OF COMPLIANCE

1. FSAR, Question 4.2

Reactor coolant system boundary materials testing complied with ASME Special Material Testing Requirements current at the construction of the Zion Station.

2. SER, page 5-4

The code current at the time of Zion construction was not adequate to establish compliance with the proposed AEC "Fracture Toughness Requirements;" however, technical specification limits were determined to allow operation.

3. Zion Station Radiological Safety Technical Specifications page 90

The fracture toughness properties of the ferritic materials in a reactor coolant pressure boundary are determined in accordance with Section III of the ASME Boiler and Pressure Vessel Code, 1974 edition.

Compliance with Section IV.A.2, 3, and IV.B are listed below. There is no Section IV - Paragraph A.4.

STATEMENT OF SECTION IV - PARAGRAPH A.2

For vessels, exclusive of bolting or other fasteners:

a. Calculated stress intensity factors shall be lower than the reference stress intensity factors by the margins specified in the ASME Code Appendix G, "Protection Against Non-Ductile Failure." The calculation procedures shall comply with the procedures specified in the ASME Code Appendix G, but additional and alternative procedures may be used if the Commission determines that they provide equivalent margins of safety against fracture, making appropriate allowance for all uncertainties in the data and analyses.

b. For nozzles, flanges, and shell regions near geometric discontinuities, the data and procedures required in addition to those specified in the ASME Code shall provide margins of safety comparable to those required for shells and heads remote from discontinuities.

c. Whenever the core is critical, the metal temperature of the reactor vessel shall be high enough to provide an adequate margin of protection against fracture, taking into account such factors as the potential for overstress and thermal shock during anticipated operational occurrences in the control of reactivity. In no case when the core is critical (other than for the purpose of low-level physics tests) shall the temperature of the reactor vessel be less than the minimum permissible temperature for the inservice system hydrostatic pressure test nor less than 40° F above that temperature required by Section IV.A.2.a.

d. If there is no fuel in the reactor during the initial preoperational system leakage and hydrostatic pressure tests, the minimum permissible test temperature shall be determined in accordance with paragraph G2410 of the ASME Code except that the factor of safety applied to each term making up the calcuated stress intensity factor may be reduced to

1.0. In no case shall the test temperature be less than $RT_{NDT}+60°$ F.

EVALUATION OF COMPLIANCE

1. WCAP-8724 and WCAP-8727

Stress intensity factors have been calculated. The requirements of the rules of ASME Appendix G have been fully satisified. The results of the analysis contained in WCAP-8724 and WCAP-8727 for Units 1 and 2, respectively, assure protection against nonductile failure of the reactor.

2. Zion Station Technical Specifications

Subparagraph C pertains to the minimum pressurization temperature with the core critical and is addressed in the Zion Station Technical Specifications, Section 3.3.2.A.

Subparagraph d is no longer applicable to Zion Station.

STATEMENT OF SECTION IV - PARAGRAPH A.3

Materials for piping, pumps, and valves shall meet the requirements of paragraph NB-2332 of the ASME Code. Materials for bolting and other fasteners shall meet the requirements of paragraph NB-2333 of the ASME Code.

EVALUATION OF COMPLIANCE

The fracture toughness properties of the ferritic materials in a reactor coolant pressure boundary are determined in accordance with Section III of the ASME Boiler and Pressure Vessel Code, 1974 edition (Zion Radiological Safety Technical Specifications, page 90).

STATEMENT OF SECTION IV - PARAGRAPH B

Reactor vessel beltline materials shall have minimum upper-shelf energy, as determined from Charpy V-notch tests on unirradiated specimens in accordance with paragraph NB-2332.2(a) of the ASME Code of 75 ft.lbs. unless it is demonstrated to the Commission by appropriate data and analysis that the lower values of upper-shelf fracture energy still provide adequate margin for deterioration from irradiation.

EVALUATION OF COMPLIANCE

The Zion RPV weld metal does not meet the minimum upper-shelf energy of 75 ft-lb. The Unit 1 and Unit 2 weld metal upper shelves have been determined to be 61.5 and 68 ft-lb, respectively. The 75 ft-lb requirement was not in effect when the Zion reactor vessels were designed and manufactured. The requirements of 10 CFR 50, Appendix G, Section V will provide assurance of adequate margins against nonductile failure.

STATEMENT OF SECTION IV - PARAGRAPH C

Reactor vessels for which the predicted value of adjusted reference temperature exceeds 200° F shall be designed to permit a thermal annealing treatment to recover material toughness properties of ferritic materials of the reactor vessel beltline.

EVALUATION OF COMPLIANCE

The adjusted reference temperature at an end-of-life fluence of $2 \times 10^{19} n/cm^2$ will exceed 200° F using predictions of Regulatory Guide 1.99 and of ASME Section III Article A-4000. The requirements to design for thermal annealing were not called for at the time the Zion RPV's were built. Annealing of the reactor vessels is feasible, if it is determined to be necessary.

SECTION V - INSERVICE REQUIREMENTS - REACTOR VESSEL BELTLINE MATERIAL

STATEMENT OF SECTION V - PARAGRAPH A

The properties of reactor vessel beltline region materials, including welds, shall be monitored by a material surveillance program conforming to the "Reactor Vessel Material Surveillance Program Requirements" set forth in Appendix H.

EVALUATION OF COMPLIANCE

See Review of 10 CFR 50, Appendix H for conformance.

STATEMENT OF SECTION V - PARAGRAPH B

Reactor vessels may continue to be operated only for that service period within which the requirements of Section IV.A.2 are satisfied, using the predicted value of the adjusted reference temperature at the end of the service period to account for the effects of irradiation on the fracture toughness of the beltline materials. The basis for the prediction shall include results from pertinent radiation effects studies in addition to the results of the surveillance program of Section V.A.

EVALUATION OF COMPLIANCE

Section 3.3.2 of the Zion Station Radiological Technical Specifications (page 79) provides limits to assure prevention of nonductile failure and requires periodic revision of operating curves based on a metal surveillance program. The bases (page 90) of these technical specifications establish that heatup and cooldown curves are based on Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition.

STATEMENT OF SECTION V - PARAGRAPH C

In the event that the requirements of Section V.B cannot be satisfied, reactor vessels may continue to be operated provided all of the following requirements are satisfied:

1. An essentially complete volumetric examination of the beltline region of the vessel including 100% of any weldments shall be made in accordance with the requirements of Section XI of the ASME Code.

2. Additional evidence of the changes in fracture toughness of the beltline materials resulting from exposure to neutron irradiation shall be obtained from results of supplemental tests, such as measurements of dynamic fracture toughness of archive material that has been subjected to accelerated irradiation.

3. A fracture analysis shall be performed that conservatively demonstrates, making appropriate allowances for all

uncertainties, the existence of adequate margins for continued operation.

EVALUATION OF COMPLIANCE

An essentially complete volumetric examination can be made using a remote operating ultrasonic tool developed by Westinghouse. The tool allows inspection from the inside surface and requires removal of vessel internals.

Evidence of material property changes due to irradiation beyond Charpy data can be obtained from fracture mechanics (Wedge Opening Loading) specimens included in some of the surveillance capsules. An NRC program to correlate irradiated fracture toughness properties of RPV steel determined from fracture mechanics specimens for comparison with Charpy results is in progress using archive material.

A fracture analysis can be performed if determined to be necessary.

STATEMENT OF SECTION V - PARAGRAPH D

If the procedures of Section V.C do not indicate the existence of an adequate safety margin, the reactor vessel beltline region shall be subjected to a thermal annealing treatment to effect recovery of material toughness properties. The degree of such recovery shall be measured by testing additional specimens that have been withdrawn from the surveillance program capsules and annealed under the same time-at-temperature conditions as those given the beltline material. The results shall provide the basis for establishment of the adjusted reference temperature after annealing. The reactor vessel may continue to be operated only for that service period within which the predicted fracture toughness of the beltline region materials satisfies the requirements of Section IV.A.2, using the values of adjusted reference temperature that include the effects of annealing and subsequent irradiation.

EVALUATION OF COMPLIANCE

Compliance with these requirements will be discussed if the need arises. Standby capsules are provided in the material surveillance programs to measure the degree of recovery achieved by annealing and to monitor the effects of subsequent irradiation.

STATEMENT OF SECTION V - PARAGRAPH E

The proposed programs for satisfying the requirements of Section V.C and V.D shall be reported to the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, for review and approval on an individual case basis at least 3 years prior to the date when the predicted fracture toughness levels will no longer satisfy the requirements of Section V.B.

EVALUATION OF COMPLIANCE

A letter has been sent to the NRC to alert them to the possibility of this event (D. L. Peoples, CECo, letter to H. R. Denton, NRC, May 6, 1980).

CONCLUSION

Zion has addressed and complies with the requirements of 10 CFR 50, Appendix G.

REFERENCES

FSAR Sections

4.0 Reactor Coolant System

FSAR Questions

Q4.2 Fracture Toughness Tests

Zion Station Radiological Safety Technical Specifications

3.3.2 and 3.3.4 Fracture Toughness Properties

Zion Station Safety Evaluation Report

5.2.3 Material Considerations

Other Reports

WCAP 8724, "ASME III, Appendix G Analysis of the Commonwealth Edison Company Zion Unit 1 Reactor Vessel.

WCAP 8727, "ASME III, Appendix G Analysis of the Commonwealth Edison Company Zion Unit 2 Reactor Vessel."

Letters

D. E. O'Brien, CECo, letter to A. Schwencer, NRC, September 7, 1977 (containing WCAP-8064 and WCAP-8132).

D. L. Peoples, CECo, letter to H. R. Denton, NRC, May 6, 1980.

APPENDIX H TO 10 CFR 50 - REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM REQUIREMENTS

Compliance with 10 CFR 50, Appendix H, is documented in the Zion Station FSAR, and the Zion Station Radiological Safety Technical Specifications. The sections of Appendix H that contain technical requirements were evaluated for compliance with those requirements.

SECTION II - SURVEILLANCE PROGRAM CRITERIA

STATEMENT OF SECTION II - PARAGRAPH A

No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods, applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E>1 MeV) at the end of the design life of the vessel will not exceed 10^{17} n/cm^2.

EVALUATION OF COMPLIANCE

Zion Station has a material surveillance program (Zion Radiological Safety Technical Specifications, page 105).

STATEMENT OF SECTION II - PARAGRAPH B

Reactor vessels constructed of ferritic materials which do not meet the conditions of Section II - Paragraph A shall have their beltline regions monitored by a surveillance program complying with the American Society for Testing and Materials (ASTM) Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels, ASTM Designation: E-185-03, except as modified by this appendix.

EVALUATION OF COMPLIANCE

The ferritic materials irradiation surveillance program is in accordance with ASTM-E-185-70, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels." The Zion Surveillance test program predates ASTM-E-185-73 (Zion Radiological Safety Technical Specifications, page 119).

The reactor vessel material surveillance program will meet the intent of the AEC Fracture Toughness Requirements for Nuclear Power Reactors 10 CFR 50 Appendix H (Zion FSAR Q4.4).

STATEMENT OF SECTION II - PARAGRAPH C.1

Surveillance specimens shall be taken from locations alongside the fracture toughness test specimens required by Section III of Appendix G. The specimen types shall comply with the requirements of Section III - Paragraph A of Appendix G (except that dropweight specimens are not required).

EVALUATION OF COMPLIANCE

1. FSAR, pages 4.5-2 and 4A-6

Surveillance capsules contain reactor vessel steel specimens taken from the shell plates located in the core region of the reactor and associated weld metal and heat affected zone metal. Fracture mechanics specimens are also taken from the core region plates and forgings, and core region weldments, including heat-affected zone material.

2. FSAR, Question 4.4

Specimen types include Charpy V-notch, tensile, and wedge opening loading specimens.

STATEMENT OF SECTION II - PARAGRAPH C2

Surveillance specimen capsules shall be located near the inside vessel wall in the beltline region, so that the specimen irradiation history duplicates to the extent practicable, within the physical constraints of the system, the neutron spectrum temperature history and maximum neutron fluence experienced by the reactor vessel inner surface. If the capsule holders are attached to the vessel wall or to the vessel cladding, construction and inservice inspection of the attachments and attachment welds shall be done according to the requirements for permanent structural attachments to reactor vessels given in the ASME Code*, Sections III and XI. The design and location of the capsules shall permit insertion of replacement capsules. Accelerated irradiation capsules may be used in addition to the required number of surveillance capsules specified in Section II Paragraph C.3. (*Defined in Section II - Paragraph A of 10 CFR 50.)

EVALUATION OF COMPLIANCE

1. FSAR, page 4.5-4

The fast neutron exposure of the specimens occurs at a faster rate than that experienced by the adjacent vessel wall, because the specimens are located between the core and the vessel. Therefore, the NDTT measurements are representative of the vessel at a later time in life.

2. WCAP-8064, page 1-2, and WCAP-8132, page 1-2

Capsule holders are attached to the thermal shield.

3. FSAR, page 4.5-2

Capsules can be replaced when the internals are removed.

Accelerated irradiation capsules are not present in the Zion Station material surveillance programs.

STATEMENT OF SECTION II - PARAGRAPHS C.3.a, C.3.b, and C.3.c

Paragraph C.3.a

For reactor vessels for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steel, making appropriate allowances for uncertainties in the measurements, that the adjusted reference temperature established in accordance with Section III - Paragraph B will not exceed 100° F at the end of the service lifetime of the reactor vessel, at least three surveillance capsules shall be provided for subsequent withdrawal as follows:

WITHDRAWAL SCHEDULE

First capsule - One-fourth service life

Second capsule - Three-fourths service life

Third capsule - Standby

In the event that the surveillance specimens exhibit, at one-quarter of the vessel's life, a shift of the reference temperature greater than originally predicted for similar material as recorded in the applicable tehnical specification, the remaining withdrawal schedule shall be modified as follows:

Second capsule - One-half service life

Third capsule - Standby

Paragraph C.3.b

For reactor vessels which do not meet the conditions of Section II - Paragraph C.3.a but for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels that the adjusted reference temperature will not exceed 200° F at the end of the service lifetime of the reactor vessel, at least four surveillance capsules shall be provided for the subsequent withdrawal as follows:

First capsule - At the time when the predicted shift of the adjusted reference temperature is approximately 50° or at one-fourth service life, whichever is earlier.

Second capsule - At approximately one-half of the time interval between first and third capsule withdrawal.

Third capsule - Three-fourths service life.

Fourth capsule - Standby.

Paragraph C.3.c

For reactor vessles which do not meet the conditions of Section II - Paragraph C.3.b, at least five surveillance capsules shall be provided for subsequent withdrawal as follows:

WITHDRAWAL SCHEDULE

First capsule - At the time when the predicted shift of the adjusted reference temperature is approximately 50° F or at one-fourth service life, whichever is earlier.

Second and third capsules - At approximately one-third and two-thirds of the time interval between first and fourth capsule withdrawal.

Fourth capsule - Three fourths of service life.

Fifth capsule - Standby.

EVALUATION OF COMPLIANCE

Zion Station is required to conform with Section II - Paragraph C.3.c as the adjusted reference temperature for both reactor vessels will exceed 200° F. The present withdrawal schedule as listed in the Zion Radiological Safety Technical Specifications does not comply with this section. However, the Technical Specifications are being updated in order to comply (Zion Radiological Safety Technical Specifications, p. 105)

STATEMENT OF SECTION II - PARAGRAPH C.3.d

Provisions shall also be made for additional surveillance tests to monitor the effects of annealing and subsequent irradiation.

EVALUATION OF COMPLIANCE

Zion Station has eight capsules per reactor vessel. Four capsules are classed as "stand-by" and can be used to monitor annealing and for further irradiation.

STATEMENT OF SECTION II - PARAGRAPHS C.3.e, C.3.f, and C.3.g

Paragraph C.3.e

Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.

Paragraph C.3.f

If accelerated irradiation capsules are employed in addition to the minimum required number of surveillance capsules, the withdrawal schedule may be modified, taking into account the test results obtained from testing of the specimens in the accelerated capsules. The proposed modified withdrawal schedule in such cases shall be approved by the Commission on an individual case basis.

Paragraph C.3.g

Proposed withdrawal schedules that differ from those specified in paragraphs a. through f. shall be submitted with a technical justification, therefor, to the Commission for approval. The proposed schedule shall not be implemented without prior Commission approval.

EVALUATION OF COMPLIANCE

The withdrawal schedules for the Zion reactor vessels are listed in the Zion Radiological Safety Technical Specifications and have not changed since the operating licenses were granted (Zion Radiological Safety Technical Specifications, page 105).

As noted in Section II - Paragraph C.3.c, this schedule is not in compliance with Appendix H and a change to the technical specification is being prepared.

STATEMENT OF SECTION II - PARAGRAPH C.4

For multiple reactors located at a single site, an integrated surveillance program may be authorized by the Commission of an individual case basis, depending on the degree of commonality and the predicted severity of irradiation.

EVALUATION OF COMPLIANCE

Zion has a separate surveillance program for each reactor vessel (Zion Radiological Safety Tecnnical Specifications, page 105).

SECTION III - FRACTURE TOUGHNESS TESTS

STATEMENT OF SECTION III - PARAGRAPH A

Fracture toughness testing of the specimens withdrawn from the capsules shall be conducted in accordance with the requirements of Section III of Appendix G, "Fracture Toughness Requirements."

STATEMENT OF SECTION III - PARAGRAPH B

The adjusted reference temperatures for the base metal, heat-affected zone, and weld metal shall be obtained from the test results by adding to the reference temperature the amount of the temperature shift in the Charpy test curves between the unirradiated material and the irradiated material, measured at the 50 foot-pound level or that measured at the 35 mil lateral expansion level, whichever temperature shift is greater. The highest adjusted reference temperature and the lowest upper-shelf energy level of all the beltline materials shall be used to verify that the fracture toughness requirements of Section V - Paragraph B of Appendix G are satisfied.

EVALUATION OF COMPLIANCE

Commonwealth Edison intends to comply with these requirements through the reactor vessel surveillance program as delineated in WCAP-8064 (Unit 1) and WCAP-8132 (Unit 2).

SECTION IV - REPORT OF TEST RESULTS

STATEMENT OF SECTION IV - PARAGRAPH A

Each capsule withdrawal and the results of the fracture thoughness tests shall be the subject of a summary technical report to be provided to the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. The report shall include a schematic diagram of the capsule locations in the reactor vessel, identification of specimens withdrawn, the test results, and the relationship of the measured results to those predicted for the reactor vessel beltline region.

STATEMENT OF SECTION IV - PARAGRAPH B

The report shall also include the dosimetry measurements performed at each specimen withdrawal, analyses of the results which yield the calculated neutron fluence which the reactor vessel beltline region has received at the time of the tests, and comparisons with the originally predicted values of fluence.

STATEMENT OF SECTION IV - PARAGRAPH C

The operating pressure and temperature limitations established for the period of operation of the reactor vessel between any two surveillance specimen withdrawals shall be specified in the report, including any changes made in operational procedures to assure meeting such temperature limitations.

EVALUATION OF COMPLIANCE

The report on the results of the tests to be performed on the capsule that was removed from Zion Station Unit 1 and from Zion Station Unit 2 following the completion of the fourth fuel cycle will be in compliance with these requirements.

CONCLUSION

Zion Station has addressed and complies with the requirements of 10 CFR 50, Appendix H.

REFERENCES

FSAR Sections

Chapter 4 Reactor Coolant System

FSAR Questions

Q4.4 Charpy V-notch Specimens

Zion Station Radiological Safety Technical Specifications

Subsection 4.3.4.D Materials Irradiation Surveillance Specimen Inspection (per unit)

Letters

D. E. O'Brien, CECO, letter to A. Schwarz, NRC, September 7, 1977 (containing WCAP-8064 and WCAP-8132).

APPENDIX I TO 10 CFR 50 - NUMERICAL GUIDES FOR DESIGN OBJECTIVES AND LIMITING CONDITIONS FOR OPERATION TO MEET THE CRITERION "AS LOW AS IS REASONABLY ACHIEVABLE" FOR RADIOACTIVE MATERIAL IN LIGHT-WATER COOLED NUCLEAR POWER REACTOR EFFLUENTS

Compliance with Appendix I, 10 CFR 50, is documented in the report, "Information Relevant To Keeping Levels of Radioactivity in Effluents To Unrestricted Areas As Low As Is Reasonably Achievable," Zion Station Units 1&2, June 4, 1976, and Amendment 1, November 12, 1976 (Appendix I Report). This report responded to the NRC questions in the D. L. Ziemann to R. L. Bolger letter, dated February 19, 1976. The information was considered sufficient by the NRC staff to determine compliance with the criteria set forth in Section II - Paragraphs A, B, and C of Appendix I. . . The sections of Appendix I that contain technical requirements were evaluated for compliance with those requirements.

SECTION II - GUIDE ON DESIGN OBJECTIVES FOR LIGHT-WATER COOLED NUCLEAR POWER REACTORS LICENSED UNDER 10 CFR 50

STATEMENT OF SECTION II - PARAGRAPH A

The calculated annual total quantity of all radioactive material above background* to be released from each light-water cooled nuclear power reactor to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways or exposure in excess of 3 millirems to the total body or 10 millirems to any organ.

*Here and elsewhere in this Appendix background means radioactive materials in the environment and in the effluents from light-water cooled power reactors not generated in, or attributed to, to the reactors of which specific account is required in determining design objectives.

EVALUATION OF COMPLIANCE

The radiological impact of radionuclides released in liquid effluents has been calculated. Consumption factors used are those given in the HERMES Computer Code and summarized in Appendix I Report, Table 1.1-2.

Specific doses are given in Appendix I Report, Table 1.1-7 for the various points of interest listed in Appendix I Report, Table 1.1-6.

All of these reported doses are within 10 CFR 50 Appendix I guidelines (Appendix I Report, page 1.1-3).

STATEMENT OF SECTION II - PARAGRAPH B.1

The calculated annual total quantity of all radioactive material above background to be released from each light-water cooled nuclear power reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 10 millirads for gamma radiation or 20 millirads for beta radiation.

EVALUATION OF COMPLIANCE

1. Appendix I Report, page 1.1-2

The radiological impact of radionuclides released in gaseous effluents has been calculated. Consumption factors used are those given in the HERMES Computer Code and summarized in Appendix I Report, Table 1.1-2. Specific doses at various points of interest are given in Appendix I Report, Tables 1.1-3 and 1.1-4. All specific doses are within Appendix I to 10 CFR 50 guidelines.

2. References 3 and 4

A discrepancy in the noble gas source term was identified in Reference 3. This was investigated and results reported to the NRC in Reference 4. The releases were well within Appendix I guidelines.

STATEMENT OF SECTION II - PARAGRAPH B.2

Notwithstanding the guidance of paragraph B.1:

(a) The Commission may specify, as guidance on design objectives, a lower quantity of radioactive material above background to be released to the atmosphere if it appears that the use of the design objectives in paragraph B.1 is likely to result in an estimated annual external dose from gaseous effluents to any individual in an unrestricted area in excess of 5 millirems to the total body; and

(b) Design objectives based upon a higher quantity of radioactive material above background to be released to the atmosphere than the quantity specified in paragraph B.1 will be deemed to meet the requirements for keeping levels of radioactive material in gaseous effluents as low as is reasonably achievable if the applicant provides reasonable assurance that the proposed higher quantity will not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 5 millirems to the total body or 15 millirems to the skin.

EVALUATION OF COMPLIANCE

The annual individual doses due to gaseous and particulate effluents are given in Appendix I Report, Table 1.1-3.

All doses are within 10 CFR 50 Appendix I guidelines.

STATEMENT OF SECTION II - PARAGRAPH C

The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form above background to be released from each light-water cooled nuclear power reactor in effluents to the atmosphere will not result in an estimated annual dose or dose commitment from such radioactive iodine and radioactive material in particulate form for any individual in an unrestricted area from all pathways of exposure in excess of 15 millirems to any organ.

EVALUATION OF COMPLIANCE

The annual doses to various organs for all gaseous and liquid effluents including iodine are given in the Appendix I Report, Tables 1.1-3 and 1.1-7.

All calculated doses are well within Appendix I to 10 CFR 50 guidelines.

STATEMENT OF SECTION II - PARAGRAPH D

In addition to the provisions of paragraphs A, B, and C above, the applicant shall include in the radwaste system all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. As an interim measure and until establishment and adoption of better values (or other appropriate criteria), the values $1000 per total body man-rem and $1000 per man-thyroid-rem (or such lesser values as may be demonstrated to be suitable in a particular case) shall be used in this cost-benefit analysis. The requirements of this paragraph D need not be complied with by persons who have filed applications for construction permits which docketed on or after January 2, 1971, and prior to June 4, 1976, if the radwaste systems and equipment described in the preliminary or final safety analysis report and amendments thereto satisfy the Guides on Design Objectives for Light-Water-Cooled Nuclear Power Reactors proposed in the Concluding Statement of Position of the Regulatory Staff in Docket-RM-50-2 dated February 20, 1974, pages 25-30, reproduced in the Annex to this Appendix I.

EVALUATION OF COMPLIANCE

Zion's application for construction permit was docketed prior to January 2, 1971. No cost-benefit analysis was requested by the NRC (Reference 5). All calculated doses are within the Appendix I of 10 CFR 50 guidelines.

SECTION III - IMPLEMENTATION

STATEMENT OF SECTION III - PARAGRAPH A.1

Conformity with the guides on design objectives of Section II shall be demonstrated by calculational procedures based upon models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated, all uncertainties being considered together. Account shall be taken of the cumulative effect of all sources and pathways within the plant contributing to the particular type of effluent being considered. For determination of design objectives in accordance with the guides of Section II, the estimation of exposure shall be made with respect to such potential land and water usage and food pathways as could actually exist during the term of plant operation: provided, that if the requirements of paragraph B of Section III are fulfilled, the applicant shall be deemed to have complied with the requirements of Paragraph C of Section II with respect to radioactive iodine if estimations of exposure are made on the basis of such food pathways and individual receptors as actually exist at the time the plant is licensed.

EVALUATION OF COMPLIANCE

1. Appendix I Report, page 1.1-2

The dispersion of airborne radionuclides and subsequent disposition in the environment have been calculated using site joint frequency meteorological data.

2. Appendix I Report, page 1.1-3

Radiation doses to man from radionuclides in liquid effluents may result from many pathways. For this station, the pathways and aquatic dispersion factors considered are shown in Table 1.1-6.

STATEMENT OF SECTION III - PARAGRAPH A.2

The charactersistcs attributed to a hypothetical receptor for the purpose of estimating internal dose commitment shall take into account reasonable deviations of individual habits from the average. The applicant may take account of any real phenomenon or factors actually affecting the estimate of radiation exposure, including the characteristics of the plant, modes of discharge of radioactive materials, physical processes tending to attenuate the quantity of radioactive material to which an individual would be exposed, and the effects of averaging exposures over times during which determining factors may fluctuate.

EVALUATION OF COMPLIANCE

1. Appendix I Report, Table 1.1-2

Consumption factors for maximum exposed individuals are listed in Appendix I Report, Table 1.1-2. These consumption factors were taken from (1) Regulatory Guide 1.109, Table A-2, (2) calculated from the HERMES computer code, and (3) from HERMES as used in Commonwealth Edison's annual and semiannual reports on station radioactive waste environmental monitoring, and occupational personnel radiation exposure.

2. Appendix I Report, Items 2.3 and 2.6

The values of X/Q and D/Q for the various points of interest are given in Item 2.3 of Appendix I Report. The calculations make use of the joint frequency distribution data as discussed in Item 2.6.

3. Appendix I Report, page 1.1-6

Radiation doses to man from radionuclides in liquid effluents may result from many pathways. For this station, the pathways and aquatic dispersion factors considered are shown in Table 1.1-6.

4. Appendix I Report, Figure 1.1-1 and Item 2.1

Schematically, the station systems affecting gaseous effluents are shown in Figure 1.1-1. More detailed P&ID's and operating parameters are given in Item 2.1.

Liquid effluents are released from the plant into a discharge stream which has an average flow of 21 x 10^5 gpm. This stream discharges into Lake Michigan.

STATEMENT OF SECTION III - PARAGRAPH B

If the applicant determines design objectives with respect to radioactive iodine on the basis of existing conditions and if potential changes in land and water usage and food pathways could result in exposures in excess of the guideline values of paragraph C of Section II, the applicant shall provide reasonable assurance that a monitoring and surveillance program will be performed to determine:

1. The quantities of radioactive iodine actually released to the atmosphere and deposited relative to those estimated in the determination of design objectives.

2. Whether changes in land and water usage and food pathways which would result in individual exposures greater than originally estimated have occurred; and

3. The content of radioactive iodine and foods involved in the changes if and when they occur.

EVALUATION OF COMPLIANCE

1. Appendix I Report, Tables 1.1-1 and 1.1-5

The release of radioactive iodine is considered in the releases of gaseous and liquid effluents as shown in Tables 1.1-1 and 1.1-5 of the Appendix I Report.

2. Appendix I Report, Tables 1.1-3, 1.1-4, and 1.1-7

The design objectives of Appendix I, 10 CFR 50 are met as shown by the results in Tables 1.1-3, 1.1-4, and 1.1-7 of the Appendix I Report.

SECTION IV - GUIDES ON TECHNICAL SPECIFICATION FOR LIMITING CONDITIONS FOR OPERATION OF LIGHT-WATER COOLED NUCLEAR POWER REACTORS LICENSED UNDER 10 CFR 50

STATEMENT OF SECTION IV - PARAGRAPH A (SUMMARY)

If the quantity of radioactive material actually released in effluents to unrestricted areas from a light-water cooled nuclear power reactor during any calendar quarter is such that the resulting radiation exposure, calculated on the same basis as the respective design objective exposure, would exceed one-half the design objective annual exposure derived pursuant to Sections II and III, the licensee shall investigate, take corrective action, and report actions to appropriate NRC Regional Office.

STATEMENT OF SECTION IV - PARAGRAPH B (SUMMARY)

The licensee shall establish an appropriate surveillance and monitoring program.

EVALUATION OF COMPLIANCE

The limiting conditions for operation and surveillance requirements of the Zion Environmental Radiological Monitoring Program are given in Sections 3.16 and 4.16 of the radiological technical specification for Zion Station Units 1&2. The reporting requirements are given in Section 6.6.3.6 of the technical specifications.

Proposed Appendix I technical specifications were submitted to NRC on February 16, 1979 (Reference 6). These specifications are still under NRC Staff review.

SECTION V - EFFECTIVE DATES

STATEMENT OF SECTION V - PARAGRAPH A

The guides for limiting conditions for operation set forth in this Appendix shall be applicable in any case in which an application was filed on or after January 2, 1971, for a permit to construct a light-water cooled nuclear power reactor.

EVALUATION OF COMPLIANCE

Not applicable to Zion, since the application was filed on July 12, 1967.

Statement of Section V - Paragraph B

For each light-water cooled nuclear power reactor constructed pursuant to a permit for which application was filed prior to January 2, 1971, the holder of the permit or a license authorizing operation of the reactor shall, within a period of twelve months from June 4, 1975, file with the Commission.

1. Such information as is necessary to evaluate the means employed for keeping levels of radioactivity in effluents to unrestricted areas as low as is reasonably achievable, including all such information as is required by Section 50.34(a), (b), and (c) not already contained in his application; and

2. Plans and proposed technical specifications developed for the purpose of keeping releases of radioactive materials to unrestricted areas during normal reactor operations, including expected operational occurrences, as low as is reasonably achievable.

EVALUATION OF COMPLIANCE

The Appendix I Report for Zion was filed with the NRC on June 4, 1976 with Amendment 1 November 12, 1976.

The revised Zion Station Technical Specifications conforming to the requirements of Appendix I and the Offsite Dose Calculation Manual (ODCM) were submitted to the NRC on February 16, 1979 (Reference 6). In addition, an informal submittal of a revised ODCM was made in December 1979. This ODCM contained changes suggested by the NRC staff during discussions on Appendix I held throughout 1979.

The discussions with the NRC staff on Appendix I Technical Specification, the ODCM, and the Environmental Monitoring Program are expected to continue in 1980.

CONCLUSION

Zion has addressed and complies with the requirements of Appendix I, 10 CFR Part 50, as documented in the Appendix I Report. The NRC staff is currently reviewing the proposed Technical Specifications for Appendix I, the Offsite Dose Calculational Manual (ODCM), and the Environmental Monitoring Program.

REFERENCES

1. "Information Relevant to Keeping Levels of Radioactivity in Effluents To Unrestricted Areas As Low As Reasonably Achievable," Zion Station Units 1&2, June 4, 1976 and Amendment 1, November 12, 1976 (Appendix I Report).

2. Radiological Safety Technical Specifications for Zion Station Units 1&2, Zion, Illinois.

   Sections 3.16 and 4.16 Environmental Radiological Monitoring Programs

   Subsection 6.6.3.b Unique Reporting Requirements - Environmental Radiological Monitoring

3. A. Schwencer, NRC, letter to R. L. Bolger, CECo, January 26, 1977.

4. R. L. Bolger, CECo, letter to A. L. Schwencer, NRC, April 26, 1977.

5. D. L. Ziemann, NRC, letter to R. L. Bolger, CECo, February 19, 1976.

6. C. Reed, CECo, letter to H. R. Denton, NRC, February 16, 1979.

APPENDIX J - REACTOR CONTAINMENT LEAKAGE TESTING FOR WATER COOLED POWER REACTORS

The Zion Technical Specifications, Paragraphs 3.10.1 and 4.10.1, contain the following requirements in regard to reactor containment leak rate testing.

> "Type A, B, and C tests of the containment shall be performed in accordance with the requirements of 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors," as published in the Federal Register, Volume 38, No. 30, February 14, 1973."

Exemptions from 10 CFR 50, Appendix J, requirements have been requested from the NRC where the Zion design does not permit full compliance.

SECTION III - LEAKAGE TESTING REQUIREMENTS

SECTION III - PARAGRAPH A - TYPE A TEST

EVALUATION OF COMPLIANCE

1. Zion Station Technical Specifications, page 212

Type A tests shall be performed with the reduced pressure test program as defined in Section III - Paragraph A.1.a of Appendix J.

a. The reference volume or absolute method of leakage rate testing shall be used for performing the test. The test will be conducted in accordance with the provisions of ANS Standard N45.5-1972.

b. The preoperational leakage rate test shall be performed at a pressure of 25 psig (P_t) followed by a second test at 47 psig (P_p).

c. The measured leakage rate Lpm shall not exceed the design basis accident leakage rate (La) of 0.1 per cent of the containment volume per 24 hours at pressure Pp.

d. The maximum allowable test leakage rate L shall be computed in accordance with Section III, Paragraph A.4.a (iii) of Appendix J.

2. Zion Station Technical Specifications, Page 213

The Type A, B, and C leakage rate tests shall be considered to be satisfactory if the acceptance criteria delineated in Appendix J, Section III, Paragraphs A.5, A.7, B.5, and C.3 are met.

3. References 7, 8, and 9

In accordance with 10 CFR 50.12, exemptions from the requirements to Section III - Paragraphs A.1.a and A.3.a were requested (References 7 and 8). The NRC approved this exemption with the stipulation that certain conservative assumptions be made in evaluating the results (Reference 10).

4. References 9, 11, 12, 14 and 15

The acceptance criteria for Type A tests for Units 1 & 2 performed in 1977, is specified in References 11 & 15. Additional correspondence concerning the Type A tests acceptance criteria is specified in References 9, 12, and 14.

SECTION III - PARAGRAPH B - TYPE B TESTS

EVALUATION OF COMPLIANCE

1. Zion Station Technical Specifications, page 213

Type B & C tests shall be performed at a pressure of 47 psig (Pp) in accordance with the provisions of Appendix J, Section III - Paragraphs B and C.

The Type A, B, and C leakage rate tests shall be considered to be satisfactory if the acceptance criteria delineated in Appendix J, Section III - Paragraphs A.5, A.7, B.5, and C.3 are met.

2. References 3 and 5

Full compliance with Section III, Paragraph B.1.b which specified techniques for conducting all Type B tests is not planned. The method used at Zion Station for determining local leak rates involves calculating the mass of air, and consequently the leak rate, which flows from a known reference volume to the test area. This technique maintains the test area at the test pressure with the reference volume temperature and pressure as variables.

SECTION III - PARAGRAPH C - TYPE C TESTS

EVALUATION OF COMPLIANCE

1. Zion Station Technical Specifications, page 213

Type B and C tests shall be performed at a pressure of 47 psig (Pp) in accordance with the provisions of Appendix J, Section III - Paragraphs B and C.

The Type A, B, and C leakage rate tests shall be considered to be satisfactory if the acceptance criteria delineated in Appendix J, Section III - Paragraphs A.B, A.7, B.5, and C.3 are met.

2. Letter from R. L. Bolger to K. R. Goller, dated May 11, 1977

Certain valves were requested to be exempted from the requirement to do Type C leakage tests on May 11, 1977 in accordance with 10 CFR 50, Section 50.12.

SECTION III - PARAGRAPH D - PERIODIC RETEST SCHEDULE

EVALUATION OF COMPLIANCE

1. Zion Station Technical Specifications, page 213

The retest schedules for Type A, B, and C tests shall be in accordance with Section III, Paragraph D of Appendix J.

2. References 3, 4, 5, 6, 7, and 9

In accordance with 10 CFR 50.12, exemptions to Section III, Paragraph D.2, which requires that air lock leakage be measured after each opening, have been requested and are undergoing NRC staff review.

SECTION IV - SPECIAL TESTING REQUIREMENTS

SECTION IV - PARAGRAPH A - CONTAINMENT MODIFICATION

EVALUATION OF COMPLIANCE

No major containment modification has been performed at Zion.

SECTION IV - PARAGRAPH B - MULTIPLE LEAKAGE BARRIER OR SUBATMOSPHERIC CONTAINMENTS

EVALUATION OF COMPLIANCE

Zion does not have a primary reactor containment barrier or a multiple barrier or subatmospheric containment. Therefore, this subsection does not apply.

SECTION V - INSPECTION AND REPORTING OF TESTS

SECTION V - PARAGRAPH A - CONTAINMENT INSPECTION AND PARAGRAPH B - REPORT OF TEST RESULTS

EVALUATION OF COMPLIANCE

1. Zion Station Technical Specifications, page 214

Inspection and reporting of tests shall be in accordance with Section V of Appendix J.

2. <u>Letter from L. D. Butterfield to R. C. DeYoung, dated February 21, 1973</u>

The results of the preoperational containment leak rate test were submitted to the NRC on February 21, 1973.

<u>CONCLUSION</u>

Zion Station has addressed and complies with the requirements of 10 CFR 50, Appendix J, to the extent possible. Where the Zion Station design, which was completed before Appendix J was published, does not allow full compliance; exemptions have been requested in accordance with 10 CFR 50.12.

<u>REFERENCES</u>

1. Zion Station Radiological Safety Technical Specifications, sections Station Units 1&2, Sections 3/4.10.1, "Containment Leakage Rate Testing."

2. L. D. Butterfield, CECO, letter to R. C. DeYoung, NRC; dated February 21, 1973.

3. B. Lee, CECo, letter to A. Giambusso, NRC, August 23, 1973.

4. R. C. DeYoung, NRC, letter to B. Lee, CECo, October 5, 1973.

5. B. Lee, CECo, letter to R. C. DeYoung, NRC, November 13, 1973.

6. L. D. Butterfield, CECo, letter to D. J. Skovholt, NRC, January 9, 1974.

7. G. J. Plimil, CECo, letter to K. R. Goller, NRC, September 26, 1975.

8. A. Schwencer, NRC, letter to R. L. Bolger, CECo, November 23, 1976.

9. D. E. O'Brien, CECo, letter to A. Schwencer, NRC, December 17, 1976.

10. D. E. O'Brien, CECo, letter to K. R. Goller, NRC, January 31, 1977.

11. A. Schwencer, NRC, letter to R. L. Bolger, CECo, January 26, 1977.

12. D. E. O'Brien, CECo, letter to A. Schwencer, NRC, February 16, 1977.

13. R. L. Bolger, CECo, letter to K. R. Goller, NRC, May 11, 1977.

14. D. E. O'Brien, CECo, letter to A. Schwencer, NRC, August 31, 1977.

15. A. Schwencer, NRC, letter to R. L. Bolger, CECo, September 17, 1977.

16. W. F. Naughton, CECo, letter to H. R. Denton, NRC, July 7, 1980.

APPENDIX K TO 10 CFR 50 - ECCS EVALUATION MODES

Compliance with 10 CFR 50 Appendix K is specifically required by 10 CFR 50.46, and Zion Station compliance is documented in Amendment No. 53 to the Facility Operating License No. DPR-39 and Amendment No. 50 to Facility Operating License No. DPR-48 for Zion Station Units 1 & 2. This amendment meets 10 CFR 50.46 criteria, and therefore 10 CFR 50 Appendix K, and was approved by the NRC as indicated in the conclusion of the Safety Evaluation Report for those amendments which states that:

"Based on the review of the submitted documents, we conclude that the results of the LOCA analysis performed with F_Q= 1.93 are conservative relative to the 10 CFR 50.46 criteria. We consider the resulting changes to the Technical Specification acceptable for operating Units 1 and 2 with a maximum of 1 percent of the steam generator tubes plugged."