YANKEE ATOMIC ELECTRIC COMPANY

Telephone 617 872-8100

YANKEE

1671 Worcester Road, Framingham, Massachusetts 01701

2.C.2.1
FYR 81-95

June 30, 1981

United States Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Division of Licensing
Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch #5

References: (a) License No. DPR-3 (Docket No. 50-29)
(b) YAEC Letter to USNRC, dated February 27, 1981, FYR 81-34

Subject: Systematic Evaluation Program Topic Assessments

Dear Sir:

Enclosed please find our assessments of the following topics:

II-2.C.	Atmospheric Transport and Diffusion Characteristics for Accident Analysis
III-4.D	Site Proximity Missiles (Including Aircraft)
VI-7.A.3	ECCS Actuation System
VI-10.A.	Testing of Reactor Trip System and Engineered Safety Features Including Response Time Testing
XV-2	Spectrum of Steam System Piping Failures Inside and Outside of Containment (PWR)
XV-3	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulatory Failure (Closed)
XV-5	Loss of Feedwater
XV-7	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
XV-9	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature
XV-10	Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentratin in the Reactor Coolant (PWR)
XV-12	Spectrum of Rod Ejection Accidents (PWR)

8107060402 810630
PDR ADOCK 05000029
P PDR

A035
S1/1

XV-14	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory
XV-15	Inadvertent Opening of a PWR Pressurizer Safety Relief Valve

We trust that you find this information satisfactory. However, if you have any questions, please contact us.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

J. A. Kay

J. A. Kay
Senior Engineer - Licensing

JAK:dad

Enclosures

YANKEE NUCLEAR POWER STATION

Topic II-2.C: Atmospheric Transport and Diffusion Characteristics for Accident Analysis

The objective of this review is to determine the appropriate on-site and near-site atmospheric transport and diffusion characteristics necessary to establish conformance with the 10CFR Part 100 Guidelines. In particular, the short-term relative ground-level air concentrations (CHI/Q) are used to estimate offsite exposures resulting from postulated accidents.

Short-term CHI/Q values for a ground-level release have been computed for various time intervals at the exclusion area boundary (EAB), a circle with a radius of 3,100 feet, and the outer boundary of the low population zone (LPZ), an approximately S-shaped boundary reflecting the fact that releases from the plant under certain meteorological conditions will remain within the valley (Topic II-1.B). Meteorological data collected onsite from January 1, 1980 through December 31, 1980 were used in the analysis.

Estimates of effluent plume dispersion and transport are complicated by the plant's location in the Deerfield River Valley whose sides rise over 800 feet above plant grade. There is evidence that the 32-foot wind sensors are often affected by localized nocturnal drainage winds flowing down the east slope of the river valley, thus biasing the lower level wind rose frequencies toward the east. As such, the 196-foot wind direction values were used to determine whether the wind flow for any given hour followed the valley or [illegible] cross-valley. The 32-foot wind speed values were used in the analysis. Vertical atmospheric stability was determined from the vertical temperature gradient between the 32-foot and 196-foot levels. Horizontal atmospheric stability was defined by fluctuations of the 196-foot horizontal wind direction (sigma theta) when winds were greater than 1.5 mps (3.3 mph) and by the vertical temperature gradient between the 32-foot and 196-foot levels when the wind speed was less than 1.5 mps.

Hourly CHI/Q values were calculated using a modified Gaussian dispersion model outline below. In order to account for the valley terrain at the site, dilution factors were calculated using a 10-sector downwind wind rose for both the EAB and LPZ. For all winds from the S clockwise through WSW cardinal wind direction sectors, it was assumed that effluents would remain in the valley. As such, winds from these four cardinal direction sectors were assumed to affect one 'upstream' downwind sector. Likewise, winds from the N clockwise through ENE cardinal wind direction sectors were also assumed to remain in the valley and affect one 'downstream' downwind sector. Winds from the other eight cardinal wind direction sectors, W clockwise through NNW and E through SSE, were assumed to be cross-valley flows which affected the E through SSE and W through NNW downwind sectors, respectively.

The procedure for determining the dilution factors for the design basis accident evaluation reflects variations in atmospheric dispersion that occur as a function of wind direction frequencies and downwind receptor distances. Dilution factors were computed for each sequential hour of measured meteorological data and for receptors positioned in each of the ten downwind sectors. These hourly CHI/Q values were calculated using a modification of the Gaussian dispersion model outlined in Regulatory Guide 1.145. Plume centerline values were used to determine the short-term dilution factors (up through 8 hours) and sector average values were used for the longer term dilution factors. The dispersion model for the plume centerline CHI/Q values considered the following effects:

1) Plume horizontal and vertical standard deviations were adjusted to account for building wake effects.

2) Lateral plume meander was allowed during periods of low wind speed and neutral and stable atmospheric conditions.

3) Lateral dispersion in the upstream and downstream downwind sectors was limited by the valley walls and included an increase in concentration due to multiple eddy reflections from the valley walls.

In addition, the sector width used to determine the hourly sector average CHI/Q values for the upstream and downstream downwind sectors was adjusted to account for the limited lateral dispersion potential due to the valley walls.

Using the hourly CHI/Q values calculated as described above, average CHI/Q values for each downwind sector were then determined for successive overlapping time intervals of 1, 2, 8, 24, 96 and 720 hours corresponding to time periods of 0 to 1 hour, 1 to 2 hours, 0 to 8 hours, 8 to 24 hours, 1 to 4 days and 4 to 30 days, respectively. For each selected downwind sector and interval size, the averaging process began with the first hourly dilution value on record and was then repeated for the same interval size starting with each subsequent hour of dispersion data. In the averaging process, the only non-zero values within a given time interval which were considered in evaluating the average dilution factor for the interval were those hours during which the wind was blowing into the downwind sector of interest. The averaged CHI/Q values were then classified into groups as a function of interval size and downwind sector, and corresponding cumulative frequency distributions of non-zero values for each group were prepared. The CHI/Q value which was exceeded 0.5% of the total time was then determined from each group, and the maximum 0.5% downwind sector value from each time interval was chosen as the design-basis CHI/Q value for that time interval.

The following CHI/Q values were determined using the above model for an assumed ground level release for the various accident time intervals at the EAB and LPZ:

Time Period	Distance & Direction	CHI/Q (sec/m^3)
0 - 1 hours	EAB (3100 feet upstream)	2.84×10^{-4}
1 - 2 hours	EAB (3100 feet downstream)	2.27×10^{-4}
0 - 8 hours	LPZ (2 miles upstream)	2.84×10^{-5}
8 - 24 hours	LPZ (6 miles downstream)	1.92×10^{-5}
24 - 96 hours	LPZ (6 miles downstream)	1.62×10^{-5}
96 - 720 hours	LPZ (6 miles downstream)	1.04×10^{-5}

We conclude that the dilution factors listed in the table above are appropriate for estimating off-site exposures resulting from postulated accidents, and that this evaluation meets the intent of current licensing practice.

YANKEE NUCLEAR POWER STATION

Topic III-4.D: Site Proximity Missiles (Including Aircraft)

The safety objective of this section is to assure that the Yankee Rowe nuclear power station is adequately protected and can be operated within an acceptable degree of safety with respect to site proximity missiles. The review was conducted in accordance with the guidance given in the USNRC Standard Review Plan's (SRP), sections 2.2.3, 3.5.1.5, and 3.5.1.6.

The scope of possible hazardous activities on the vicinity of the Yankee Rowe plant has been discussed in SEP Topic II-1.C, "Potential Hazards Due to Nearby Industrial, Transportation, Institutional and Military Facilities." As indicated, there is minimal industrial activity in the plant vicinity. The separation distance and valley terrain between the plant and any industrial facilities, highways, railroads, gas pipelines, or military facilities is such that the risk associated with potential missiles from these concerns are well within the SRP 2.2.3 guidelines.

In addition to the review of fixed facilities and ground transportation routes in the site area, the potential of aircraft accident generated missiles has also been evaluated in detail. The methodology employed in this analysis is the same as that outlined in SRP 3.5.1.6.

There are four airports within thirty miles of Yankee Rowe: 1) Harriman-and West (North Adams), 2) Bennington State, 3) Pittsfield, and 4) Turners Falls. Each can be described in general as being small airports typically handling light, single-engine, private aircraft. The location and description of each airport, along with an estimate of the number of annual operations (take-offs and landings), is contained on FAA Forms 5010-1 in Appendix A.

Table 1 summarizes each airport's annual operations. As shown, none of the airports are within ten miles of Yankee, and all of the airport's reported annual operations are well within the 1,000 times distance squared criteria of SRP 3.5.1.6 which, if exceeded, would indicate the possible need for further analysis of aircraft from these airports effecting plant operations.

TABLE 1

	Distance (d) Statute Miles	$1000 \times d^2$	Annual Number of Operations
North Adams	12.0	144000	32500
Bennington	19.0	361000	10950 (a)
Pittsfield	28.0	784000	50000
Turners Falls	22.0	484000	34873

(a) Annual operation information obtained directly from Bennington Airport Manager, April 1, 1981.

In addition to the above noted airports, there are two federal airways, V2-14 and J16-94, which could bring aircraft near the plant site. V2-14 is used by aircraft below 18,000 feet, whereas J16-94 is used by aircraft at altitudes of 18,000 feet and above. Both airways have a total width of 8 nautical miles (9.2 statute miles); i.e. 4 nautical miles each side of centerline. Yankee is located approximately 2.5 nautical miles north of the V2-16 centerline, and 5 nautical miles north of the J16-94 centerline. In estimating the annual number of aircraft that may pass near the site due to these airways, a count was made of the IFR traffic on both these airways for the "peak traffic day" of 1980 (August 22). Federal Aviation Administration radar records associated with these corridors indicate a total of 155 aircraft could have flown near the plant site during the peak day. Based upon the peak day traffic, an annual estimate of 56,575 aircraft passing near the site was calculated. Employing the analytical model given in SRP 3.5.1.6, it is calculated on a conservative basis that the overall probability of an aircraft associated with these air corridors striking the plant is approximately 1.4 x 10^{-7} per year. This is an acceptable level of risk in accordance with the acceptance criteria of SRP 2.2.3.

In calculating the risk probability, an effective plant area of 0.0075 square miles was used. This was determined by assuming an aircraft crash angle of 30 degrees relative to the principal plant structures, including non-safety related buildings attached to the plant. Since Yankee Rowe is located in a valley, the crash angle was based on the kinds of aircraft identified within the V2-14 and J16-94 airways, and the width of the valley and height of the mountains surrounding the plant. An inflight crash rate of 3 x 10^{-9} per aircraft mile was used in the calculation (SRP 3.5.1.6). No information was identified from the FAA on future growth of traffic in these corridors. However, since the calculated probability of 1.4 x 10^{-7} conservatively assumes that a single day peak traffic load in the corridors is maintained throughout the year, any future real growth in aircraft activities in these corridors over the remainder of plant life would not be expected to change significantly the calculated risk factor.

We conclude that the risk of missile impacts (including aircraft) on the Yankee Rowe plant from offsite sources is acceptably low within the criteria of SRP 2.2.3.

FEDERAL AVIATION ADMINISTRATION
AIRPORT MASTER RECORD

11 AIRSPACE ANALYSIS DETERMINATION •NOT ANALYZED
5 REG. ANE DIST.
09127.A

8 ASSOC. CITY NORTH ADAMS
9 STATE CITY IN MA
32 AERO CHART NEW YORK
2 ARPT LCTN ID 2B6
10 OFFICIAL AIRPORT NAME HARRIMAN-AND-WEST
12 STATE NEAREST CITY IN MA
30 ARPT ACREAGE 130E
11 NEAREST CITY NORTH ADAMS
14 STATE ARPT IN MA
ARPT OPEN TO PUBLIC Y
13 COUNTY AIRPORT IN BERKSHIRE
28 MAG VARN
NASP FED AGREEMENTS NGY
15 OWNER CITY OF NORTH ADAMS
16B CITY ST ZIP NORTH ADAMS, MA 01247
ADDRESS 16A ST BOX CITY HALL
☒ AIRPORT ☐ SEAPLANE BASE ☐ HELIPORT ☐ STOLPORT
17 ☒ PUBLIC OWNED ☐ PRIVATE OWNED
20 PHONE NO (with area code) 413 664-4585
18 MANAGER WM FLTEL
19B CITY ST ZIP NORTH ADAMS MA 01247
ADDRESS 19A ST BOX NORTH ADAMS MUNI ARPT

CITY OF NORTH ADAMS
HARRIMAN-AND-WEST AIRPORT
CITY HALL
NORTH ADAMS, MA 01247

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2. Your FSS is ALBANY
4. FSS on airport NO
3. For a toll free call to your FSS, call 1-800-833-4509 If a toll free number is not printed, call 518-869-9225

21 DISTANCE AND DIRECTION TO AIRPORT FROM ASSOCIATED CITY: NO. STATUTE MILES 03 DIRECTION W
23 AIRPORT LATITUDE 42-41-45N
24 LONGITUDE 073-10-15W
E
25 AFP N
26 ELEVATION 00654 S

33 RUNWAY	34 PHYS LENGTH	35 WIDTH	36 SURFACE COMPOSITION	FAA GROSS WT STRENGTH (1000 LBS) 37 SNGL WHL	38 DUAL WHL	39A DUAL TAND	39B DOUBLE DUAL TAND	40 % EFF GRADIENT	42 MARKING TYPE & CONDITION	43 C/L LGTS	45 RWY LGTS INTENS
11/29	4300	100	ASPH	25				.15	BASIC-F	N	M

RUNWAY END	46 ALIGNMENT TRUE	47 Hi Spd Exit	48 APPROACH	49 TD ZONE	50 REIL	51 APCH SLOPE INDCR	52 RVR	53 RVV	54 CTLG OBSTRUCTION	55 DSTC FM THRESH	56 APCH RATIO	57 CLOSE IN OBSTN	58 RGT TFC	60 BLAST FENCE
11				N	N	NONE	N	N	TREES	800	20-1	N	N	N
29		N		N	N	NONE	N	N	TREES	200	• 1-1	•Y	N	Y

RUNWAY END	60 ARRESTING DEVICE	61 THRESHOLD DISPLACED	LANDING LENGTH 62 DAY	63 NIGHT	TAKEOFF LENGTH 64 DAY	65 NIGHT	THRESHOLD 68 LATITUDE	69 LONGITUDE	70 ELEV
11									
29	I	800							

71 STANDARD TIME ZONE +05
72 DAYLIGHT SAVING TIME Y
75 AIRPORT BEACON LIGHT
NONE ☐ CLEAR & GREEN ☐
CLEAR ☒ CLEAR & YELLOW ☐
76 BCN OPER SCHEDULE DUSK-DAWN
73 RWY LGT OPER SKED DUSK-DAWN
77 LANDING FEE FOR NON-COMMERCIAL USERS N
79 CUSTOMS AIRPORT OF ENTRY N
80 CUSTOMS LANDING RIGHTS AIRPORT N

74 AIRPORT ATTENDANCE SCHEDULE MONTHS	DAYS	HOURS
ALL	ALL	DAWN-DUSK

ENGINE OIL: 100 PISTON Y; 101 TURBINE Y
REPAIRS: AIRFRAME MAJOR 102 Y, MINOR 103 Y; POWER PLANT MAJOR 104 Y, MINOR 105 Y
OXYGEN: 106 HI PRESSURE Y; 107 LO PRESSURE Y
OXY REPL BOTTLES: 108 HI PRESSURE Y; 109 LO PRESSURE Y
110 AUX POWER FOR OPNL FAC
114 F-TYPE FUEL FOR PUBLIC SALE
12 ☐ 18 ☐ 22 ☐ 30 ☐ •
34 ☐ 35 ☐ 40 ☐ 45 ☐
81 TAXIWAY LIGHTS N
82 WIND INDICATOR Y
83 SEGMENTED CIRCLE Y
93 HELIPAD ON AIRPORT N
94 WEATHER STA ON ARPT N
95 ADMIN TRML BLDG Y
97 CARGO HANDLING
98 NO. T-HANGARS 04
99 NO. OTHER HNGRS 05
115 MIL JOINT USE AGREEMENT N
116 MIL LANDING RIGHTS Y
117 BASED MIL & TYPE OPERS
118 TYPE & NO. MILITARY AIRCRAFT NONE
119 ESTD TOTAL ANNUAL MIL OPERS 0
120 ESTD ANNUAL AIR TAXI OPERS 2000

ESTIMATED GENERAL AVIATION OPERATIONS 121 ANNUAL LOCAL	122 ANNUAL ITINERANT	123 TOTAL PEAK MONTH
22000	10500	3000

BASED GEN AVIATION ACFT
SINGLE ENGINE
124 4 PLACE & OVER 15
125 UNDER 4 PLACE 5
126 MULTI-ENG 6
127 HELICOPTER • 0
128 SEAPLANE 0

134 FAR 139 CFR INDEX N
145 UNICOM FREQ 122.8
142 TFC PATTERN ALT
SEAPLANE BASE DATA: 129 SHELTERED 130 MOORING BUOYS 131 DOCK 132 RAMP 133 BEACHING GEAR

137 REMARKS
E056 RWY 29 APCH RATIO 20-1 FROM DSPLCD THR.
E057 RWY 29 +8' BLAST FENCE 75' OUT FROM PAVEMENT END.
E111 FILED FOR RECORD PURPOSES ONLY.
E114 FUEL CODES: 100, A.
E127 10 BASED GLIDERS
E137-01 MTN RANGES 9100' N AT ELEV 1894', 4300' N AT ELEV 2254', 10000'
S AT ELEV 2951', 12000' S AT ELEV 2698'.
E137-02 EXTENSIVE GLIDER OPNS DAWN-DUSK SFC 10000 YEAR ROUND;PRIMARILY
WEEKENDS & HOLIDAYS.

135 LATEST INSP OR INFO REQ 20 NOV 1979
136 INSPN Y

NOTE: Except on #22 & 23, Y = Yes, N = No. "•" = See Remarks. Re #52, 1 = Approach, 2 = Approach and Roll Out, 3 = Approach, Mid-Point, and Roll Out. Re #45, L = Low, M = Medium, H = High.
Abbreviations are listed in FAA Contractions Handbook, Order 7340 1D

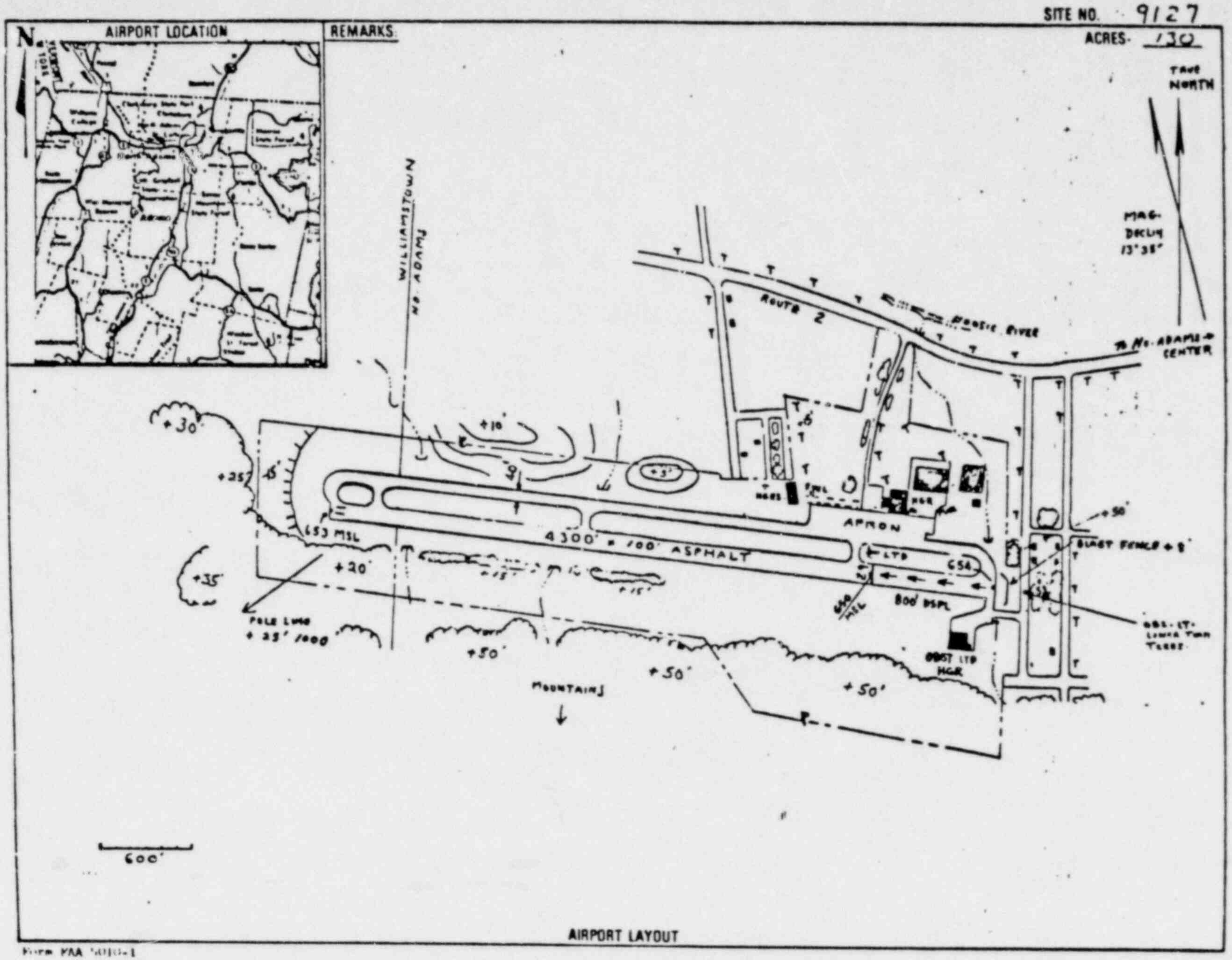
FAA Form 5010-1 (2-78)

SUPERSEDES PREVIOUS EDITION

APPENDIX A (Page 1 of 8)
HARRIMAN-AND-WEST AIRPORT

SITE NO. 9127

ACRES: 130



Form FAA 5010-1

APPENDIX A (Page 2 of 8)

HARRIMAN-AND-WEST AIRPORT

SEE REVERSE SIDE FOR INSTRUCTIONS

DEPARTMENT OF TRANSPORTATION
FEDERAL AVIATION ADMINISTRATION

AIRPORT MASTER RECORD

111 AIRSPACE ANALYSIS DETERMIN. •NOT ANALYZED
DATE OF PRINT 10/26/79
6 REG. ANE DIST.
1 SITE NO. 25361.A
8 ASSOC. CITY BENNINGTON
9 STATE CITY IN VT
3 AERO CHART NEW YORK
2 ARPT LCTN ID 5B5
10 OFFICIAL AIRPORT NAME BENNINGTON STATE
11 NEAREST CITY OLD BENNINGTON
12 STATE NEAREST CITY IN VT
30 ARPT ACREAGE 100E
13 COUNTY AIRPORT IN BENNINGTON
14 STATE ARPT IN VT
96 ARPT OPEN TO PUBLIC Y
28 MAG VARN
22 NASP FED AGREEMENTS NGY
15 OWNER STATE OF VERMONT
ADDRESS 16A STATE HOUSE 133 STATE ST
16B CITY, ST, ZIP MONTPELIER, VT 05602
17 ☒ PUBLIC OWNED ☐ PRIVATE OWNED
☒ AIRPORT ☐ SEAPLANE BASE ☐ HELIPORT ☐ STOLPORT
20 PHONE NO. (with area code) 802-442-6328
18 MANAGER DAVID COREY
ADDRESS 18A BENNINGTON STATE ARPT
18B CITY, ST, ZIP BENNINGTON, VT 05201

STATE OF VERMONT
BENNINGTON STATE AIRPORT
STATE HOUSE 133 STATE ST
MONTPELIER, VT 05602

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2. Your FSS is ALBANY
4. FSS on airport NO
3. For a toll free call to your FSS, call 1-800-833-4509 If a toll free number is not printed, call - 518-869-9225

21 DISTANCE AND DIRECTION TO AIRPORT FROM ASSOCIATED CITY NO. STATUTE MILES 04 DIRECTION W
23 AIRPORT LATITUDE 42-54-00N
24 LONGITUDE 073-15-00W E
25 ARP N
26 ELEVATION 00821 S

33 RUNWAY	34 PHYS LENGTH	35 WIDTH	36 SURFACE COMPOSITION	FAA GROSS WT STRENGTH (1000 LBS) 37 SNGL WHL	38 DUAL WHL	39A DUAL TAND	39B DOUBLE DUAL TAND	40 % EFF GRADIENT	42 MARKING TYPE & CONDITION	43 C/L LGTS	45 RWY LGTS INTENS
13/31	3200	60	ASPH	12.5				.04	BASIC-G	N	M

RUNWAY END	46 ALIGNMENT TRUE	LIGHTING 47	48 APPROACH	49 TD ZONE	50 REIL	51 APCH SLOPE INDCR	52 RVR	53 RVV	APPROACH SURFACE DATA 54 CTLG OBSTRUCTION	55 DSTC FM THRESH	56 APCH RATIO	57 CLOSE IN OBSTN	58 RGT TFC	59 BLAST FENCE
13		N		N	•Y	•V2-L	N	N	TREES	3500	20-1	N	N	N
31		N		N	N	NONE	N	N	GRND	000	23-1	N	Y	N

RUNWAY END	60 ARRESTING DEVICE	THRESHOLD DISPLACED	LANDING LENGTH DAY	LANDING LENGTH NIGHT	TAKEOFF LENGTH DAY	TAKEOFF LENGTH NIGHT	THRESHOLD LATITUDE	THRESHOLD LONGITUDE	70 ELEV
13									
31									

71 STANDARD TIME ZONE •E5
72 DAYLIGHT SAVING TIME Y
75 AIRPORT BEACON LIGHT •
NONE ☐ CLEAR & GREEN ☒
CLEAR ☐ CLEAR & YELLOW ☐
76 BCN OPER SCHEDULE DUSK-DAWN
73 RWY LGT OPER SKED •
77 LANDING FEE FOR NON-COMMERCIAL USERS N
79 CUSTOMS AIRPORT OF ENTRY N
80 CUSTOMS LANDING RIGHTS AIRPORT N

74 AIRPORT ATTENDANCE SCHEDULE

MONTHS	DAYS	HOURS
ALL	ALL	0900-DUSK

ENGINE OIL 100 PISTON Y
101 TURBINE N
REPAIRS MAJOR / MINOR
AIRFRAME 102 Y 103 Y
POWER PLANT 104 Y 105 Y
OXYGEN 106 HI PRESSURE N
107 LO PRESSURE N
OXY REPL BOTTLES 108 HI PRESSURE N
109 LO PRESSURE N
110 AUX POWER FOR OPNL FAC
114 F-TYPE FUEL FOR PUBLIC SALE •
12 ☐ 18 ☐ 22 ☐ 30 ☐
34 ☐ 35 ☐ 40 ☐ 45 ☐
81 TAXIWAY LIGHTS Y
82 WIND INDICATOR Y
83 SEGMENTED CIRCLE Y
93 HELIPAD ON AIRPORT N
94 WEATHER STA ON ARPT •Y
95 ADMIN TRML BLDG Y
97 CARGO HANDLING
98 NO. T-HANGARS 11
99 NO. OTHER HNGRS 2
115 MIL JOINT USE AGREEMENT N
116 MIL LANDING RIGHTS Y
117 BASED MIL & TYPE OPERS
118 TYPE & NO. MILITARY AIRCRAFT NONE
119 ESTD TOTAL ANNUAL MIL OPERS
120 ESTD ANNUAL AIR TAXI OPERS

ESTIMATED GENERAL AVIATION OPERATIONS

121 ANNUAL LOCAL	122 ANNUAL ITINERANT	123 TOTAL PEAK MONTH
10950		

BASED GEN AVIATION ACFT
SINGLE ENGINE
124 4 PLACE & OVER 10
125 UNDER 4 PLACE 8
126 MULTI-ENG 5
127 HELICOPTER 0
128 SEAPLANE 1

134 FAR 139 CFR INDEX EU 12/74
145 UNICOM FREQ 122.8
142 TFC PATTERN ALT
SEAPLANE BASE DATA 129 SHELTERED 130 MOORING BUOYS 131 DOCK 132 RAMP 133 BEACHING GEAR

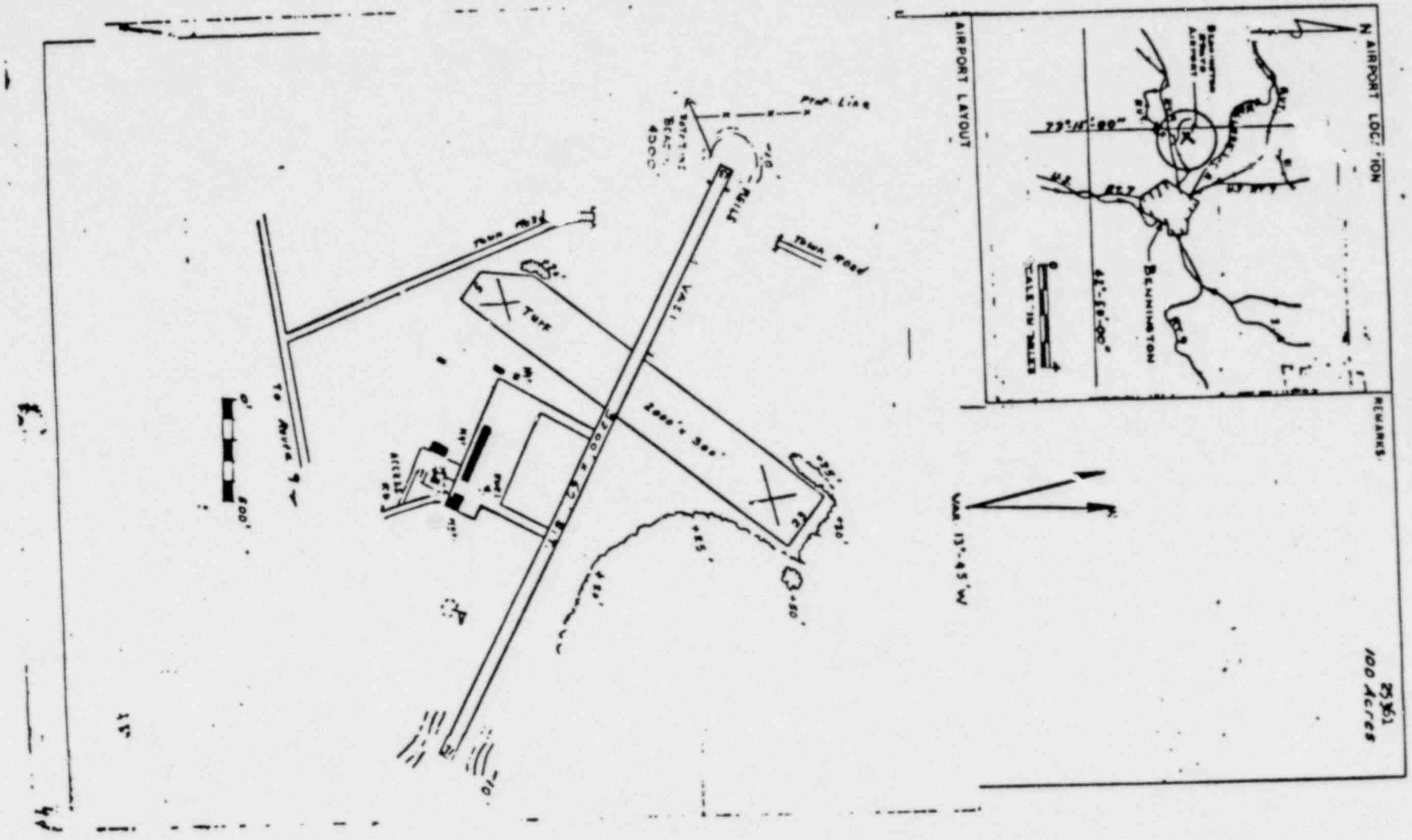
137 REMARKS
E050 •AFTER RWY LIGHTS ARE ON..KEY 122.8 3 TIMES FOR REILS.
E058 RWY 13 UPWIND OF RWY 13 END 150'.
E051 RWY 13 VASI LOCATED 400FT & 900FT FM RWY END.
E073 •FOR RWY LIGHTS KEY 122.8 5 TIMES.
E075 LOCATED LAT.42-53-31, LONG.. 73-16-01. ELEV 1330MSL/TOP OF BEACON/
E094 SAWRS ONLY.
E111 EXISTED PRIOR TO 1959.
E114 FUEL CODES: 100.

135 LATEST INSP OR INFO REQ 25 SEP 1979
136 INSPN Y

NOTE Except in #22 & 23, Y = Yes, N = No, "•" - See Remarks. Re #52, 1 = Approach, 2 = Approach and Roll Out, 3 = Approach, Mid-Point, and Roll Out. Re #45, L = Low, M = Medium, H = High.
Abbreviations are listed in FAA Contractions Handbook, Order 7340.1D

FAA Form 5010-1 (6-75) SUPERSEDES PREVIOUS EDITION

APPENDIX A (Page 3 of 8)
BENNINGTON STATE AIRPORT



APPENDIX A (Page 4 of 8)
BENNINGTON STATE AIRPORT

FORM APPROVED, OMB No. 04-R0051

DEPARTMENT OF TRANSPORTATION
FEDERAL AVIATION ADMINISTRATION
AIRPORT MASTER RECORD

SEE REVERSE SIDE FOR INSTRUCTIONS

111 AIRSPACE ANALYSIS DETERMIN. *NOT ANALYZED
DATE OF PRINT 10/10/79
5 REG. ANE DIST.
1 SITE NO. 09203.A

8 ASSOC. CITY PITTSFIELD
9 STATE CITY IN MA
32 AERO CHART NEW YORK
10 OFFICIAL AIRPORT NAME PITTSFIELD MUNI
2 ARPT LCTN ID PSF
11 NEAREST CITY PITTSFIELD
12 STATE NEAREST CITY IN MA
30 ARPT ACREAGE 281E
13 COUNTY AIRPORT IN BERKSHIRE
14 STATE ARPT IN MA
15 ARPT OPEN TO PUBLIC Y
16 OWNER CITY OF PITTSFIELD
28 MAG VARN
22 NASP/FED AGREEMENTS NGY
ADDRESS 16A ST/BOX/RD
16B CITY/ST/ZIP PITTSFIELD MA 01201
17 ☒ PUBLIC OWNED ☐ PRIVATE OWNED ☒ AIRPORT ☐ SEAPLANE BASE ☐ HELIPORT ☐ STOLPORT
18 MANAGER MAHLON RICHARDS
20 PHONE NO. (with area code) 413-499-0213
ADDRESS 19A ST/BOX/RD PITTSFIELD MUNICI
19B CITY/ST/ZIP PITTSFIELD MA 01201

CITY OF PITTSFIELD
PITTSFIELD MUNI AIRPORT
PITTSFIELD MA 01201

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4. FSS on airport NO
3. For a toll free call to your FSS, call 1-800-833-4589 If a toll free number is not printed, call 518-869-9225

21 DISTANCE AND DIRECTION TO AIRPORT FROM ASSOCIATED CITY NO. STATUTE MILES 02 DIRECTION W
23 AIRPORT LATITUDE 42-25-36.3N
24 LONGITUDE 073-17-36.2W S
25 AMP Y
26 ELEVATION 1194 S

33 RUNWAY	34 PHYS LENGTH	35 WIDTH	36 SURFACE COMPOSITION	FAA GROSS WT STRENGTH (1000 LBS) 37 SNGL WHL	38 DUAL WHL	38A DUAL TAND	38B DOUBLE DUAL TAND	40 % EFF GRADIENT	47 MARKING TYPE & CONDITION	43 C.A. LGTS	45 RWY LGTS INTNS
08/26	5000	100	ASPH	44	54			1.00	BASIC-G	N	M
14/32	3500	100	ASPH	25				.80	BASIC-F	N	M

RUNWAY END	46 ALIGNMENT TRUE	LIGHTING 47 Hi Speed Exit	48 APPROACH	49 TD ZONE	50 REIL	51 APCH SLOPE INDCR	52 RVR	53 RVV	APPROACH SURFACE DATA 54 CTLG OBSTRUCTION	55 OBSTC FM THRESH	56 APCH RATIO	57 CLOSE IN OBSTN	58 RGT TFC	59 BLAST FENCE
08	064	N		N	N	NONE	N	N	TREES	1550	22-1	N	N	N
26	244	N	*MALSF	N	N	NONE	N	N	TREES	2100	28-1	N	Y	N
14	122	N		N	N	NONE	N	N	TREES	200	1-1	Y	N	N
32	302	N		N	N	NONE	N	N	TREES	600	13-1	N	Y	N

RUNWAY END	ARRESTING 60 DEVICE (For End)	61 THRESHOLD DISPLACED	LANDING LENGTH 62 DAY	63 NIGHT	TAKEOFF LENGTH 64 DAY	65 NIGHT	THRESHOLD 68 LATITUDE	69 LONGITUDE	70 ELEV
08							42-25-25.331	073-18-10.338S	1194
26							42-25-46.779	073-17-10.285S	1144
14							42-25-46.117	073-17-49.835S	1170
32							42-25-27.399	073-17-10.682S	1142

71 STANDARD TIME ZONE *05
72 DAYLIGHT SAVING TIME Y
75 AIRPORT BEACON LIGHT Y
NONE ☐ CLEAR & GREEN ☒
CLEAR ☐ CLEAR & YELLOW ☐
76 BCN OPER SCHEDULE DUSK-DAWN
73 RWY LGT OPER SKED DUSK-DAWN
77 LANDING FEE FOR NON-COMMERCIAL USERS N
79 CUSTOMS AIRPORT OF ENTRY N
80 CUSTOMS LANDING RIGHTS AIRPORT N

74 AIRPORT ATTENDANCE SCHEDULE

MONTHS	DAYS	HOURS
ALL	ALL	0700-2200

ENGINE OIL: 100 PISTON Y; 101 TURBINE Y
REPAIRS: AIRFRAME MAJOR 102 Y, MINOR 103 Y; POWER PLANT MAJOR 104 Y, MINOR 105 Y
OXYGEN: 106 HI PRESSURE N; 107 LO PRESSURE N
OXY REPL BOTTLES: 108 HI PRESSURE N; 109 LO PRESSURE N
110 AUX POWER FOR OPRNL FAC
114 F-TYPE FUEL PUMP PUBLIC SALE
12 ☐ 18 ☐ 22 ☐ 30 ☐
34 ☐ 35 ☐ 40 ☐ 45 ☐

81 TAXIWAY LIGHTS N
82 WIND INDICATOR Y
83 SEGMENTED CIRCLE N
93 HELIPAD ON AIRPORT N
94 WEATHERSTA ON ARPT *N
95 ADMIN TRML BLDG Y
97 CARGO HANDLING
98 NO. T-HANGARS 11
99 NO. OTHER HNGRS 5
115 MIL JOINT USE AGREEMENT N
116 MIL LANDING RIGHTS Y
117 BASED MIL & TYPE OPERS
118 TYPE & NO. MILITARY AIRCRAFT NONE
119 ESTD TOTAL ANNUAL MIL OPERS 75
120 ESTD ANNUAL AIR TAXI OPERS * 6000
ESTIMATED GENERAL AVIATION OPERATIONS
121 ANNUAL LOCAL 35000
122 ANNUAL ITINERANT 15000
123 TOTAL PEAK MONTH 6000
134 FAR 139 CFR INDEX LD 1277*
145 UNICOM FREQ 122.7
142 TFC PATTERN ALT

BASED GEN AVIATION ACFT
SINGLE ENGINE
124 4 PLACE & OVER 24
125 UNDER 4 PLACE 10
126 MULTI-ENG 6
127 HELICOPTER 0
128 SEAPLANE 0

SEAPLANE BASE DATA 129 SHELTERED 130 MOORING BUOYS 131 DOCK 132 RAMP 133 BEACHING GEAR

137 REMARKS
E048 RWY 26 CFOR MALSF RWY 26 KEY 121.85 5 TIMES. 7 TIMES FOR OFF.
E075 ROTG BCN LCTD AT LAT 42-25-22.7N & LONG 073-17-27.5W.
E094 SAWRS ONLY.
E111 BUILT PRIOR TO 1959
E114 FUEL CODES: A, 100LL.
E120 SCHEDULED AIR TAXI AVAILABLE BY PRECISION AIRLINES.
E137-01 OBSTN LGTD POLES 6500' E AT 1446' MSL, 5500' SE AT 1575' MSL, 5500' S AT 1860' MSL, 10200' SW AT 2166' MSL, 14100' NW AT 1889' MSL.
E137-02 CONSISTS OF 9 BARS AT SEVEN STATIONS. THE THREE OUTERMOST STATIONS HAVE SEQUENCE FLASHING LGTS.
E137-03 RWY [illegible]/32 CLSD TO ACR NGT OPS.

AIM 3 135 LATEST INSP OR INFO MO 18 SEP 1979 136 INSPN Y

NOTE: [illegible]

FAA Form 5010-1 [illegible]
SUPERSEDES PREVIOUS EDITION

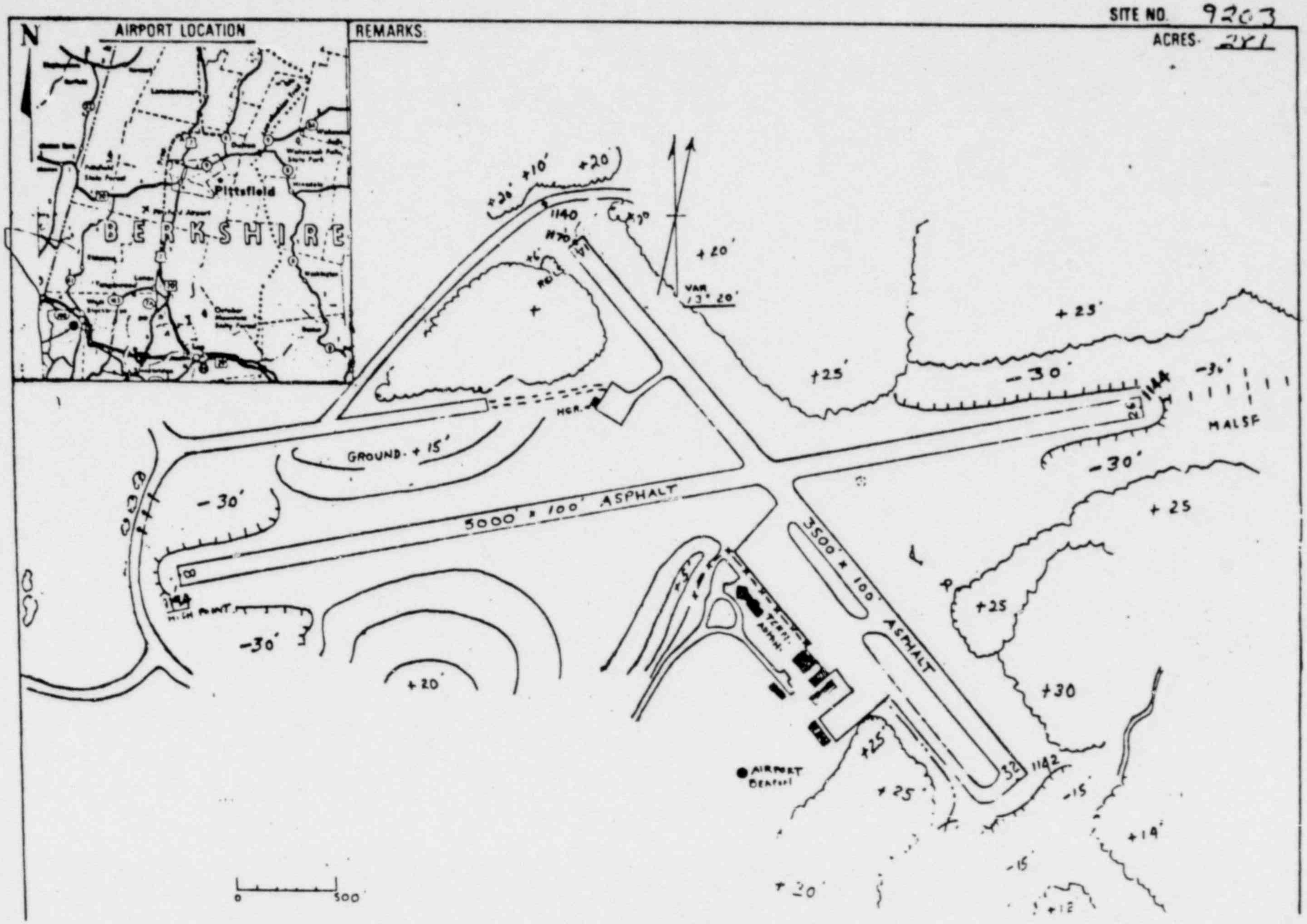
APPENDIX A (Page 5 of 8)

PITTSFIELD AIRPORT

SITE NO. 9203

ACRES. 281

REMARKS:



APPENDIX A (Page 6 of 8)
PITTSFIELD AIRPORT

FEDERAL AVIATION ADMINISTRATION

AIRPORT MASTER RECORD

111 AIRSPACE ANALYSIS DETERMN. •NOT ANALYZED	DATE OF PRINT 11/30/78	5 REG. ANE DIST.	1 SITE NO. 09090.A
8 ASSOC. CITY MONTAGUE	9 STATE CITY IN MA	32 AERO CHART NEW YORK	
10 OFFICIAL AIRPORT NAME TURNERS FALLS		2 ARPT LCTN ID 0B5	
11 NEAREST CITY TURNERS FALLS	12 STATE NEAREST CITY IN MA	30 ARPT ACREAGE 2270	
13 COUNTY AIRPORT IN FRANKLIN	14 STATE ARPT IN MA	16 ARPT OPEN TO PUBLIC Y	
15 OWNER TOWN OF MONTAGUE	28 MAG VARN 1425W	22 NASP/FED AGREEMENTS NG	
ADDRESS: 16A ST/BOX/RT MONTAGUE	16B CITY/ST/ZIP MASS 01351		
17 ☒ PUBLIC OWNED ☐ PRIVATE OWNED	☒ AIRPORT ☐ SEAPLANE BASE	☐ HELIPORT	☐ STOLPORT
18 MANAGER CHARLES BOHONOWICZ	20 PHONE NO. (with area code) 413-863-9391		
ADDRESS: 19A ST/BOX/RT TURNERS FALLS ARPT	19B CITY/ST/ZIP TURNERS FALLS, MA 01376		

TOWN OF MONTAGUE
TURNERS FALLS AIRPORT
MONTAGUE
MASS 01351

Between FAA contacts, please advise your Flight Service Station (FSS) of changes to items preceded by a dot (•). 3. Interphone between FSS & arpt. NO
2. Your FSS is WINDSOR LOCKS
4. FSS on airport NO
3. For a toll free call to your FSS, call 800-243-0310 If a toll free number is not printed, call 203-623-2416

21 DISTANCE AND DIRECTION TO AIRPORT FROM ASSOCIATED CITY: NO. STATUTE MILES 04 DIRECTION N

23 AIRPORT LATITUDE 42-35-30N 24 LONGITUDE 072-31-25W E 25 ARP Y 26 ELEVATION 00356 S

33 RUNWAY	34 PHYS LENGTH	35 WIDTH	36 SURFACE COMPOSITION	FAA GROSS WT STRENGTH (1000 LBS) 37 SNGL WHL	38 DUAL WHL	39A DUAL TAND	39B DOUBLE DUAL TAND	40 % EFF GRADIENT	42 MARKING TYPE & CONDITION	43 CA LGTS	45 RWY LGTS INTENS
16/34	3000	75	ASPH	30				.35	BASIC-G	N	L

RUNWAY END	46 ALIGNMENT TRUE	LIGHTING 47 Hi Speed Exit	48 APPROACH	49 TD ZONE	50 REIL	51 APCH SLOPE INDCR	52 RVR	53 RVV	APPROACH SURFACE DATA 54 CTLG OBSTRUCTION	55 DSTC FM THRESH	56 APCH RATIO	57 CLOSE IN OBSTN	58 RGT TFC	59 BLAST FENCE
16		N		N	N	NONE	N	N	TREES	700	21-1	N	N	N
34		N		N	N	NONE	N	N	TREES	800	17-1	N	N	N

RUNWAY END	60 ARRESTING DEVICE (For End)	61 THRESHOLD DISPLACED	LANDING LENGTH 62 DAY	63 NIGHT	TAKEOFF LENGTH 64 DAY	65 NIGHT	THRESHOLD 68 LATITUDE	69 LONGITUDE	70 ELEV
16									
34									

71 STANDARD TIME ZONE •E5	76 BCN OPER SCHEDULE DUSK-DAWN	73 RWY LGT OPER SKED DUSK-DAWN	74 AIRPORT ATTENDANCE SCHEDULE MONTHS ALL DAYS ALL HOURS 0800-2000
72 DAYLIGHT SAVING TIME Y	77 LANDING FEE FOR NON-COMMERCIAL USERS N		
75 AIRPORT BEACON LIGHT NONE ☐ CLEAR & GREEN ☒ CLEAR ☐ CLEAR & YELLOW ☐	79 CUSTOMS AIRPORT OF ENTRY N	80 CUSTOMS LANDING RIGHTS AIRPORT N	
ENGINE OIL 100 PISTON Y 101 TURBINE N	81 TAXIWAY LIGHTS •N	97 CARGO HANDLING	98 NO. T-HANGARS 0 99 NO. OTHER HNGRS 3
	82 WIND INDICATOR Y	115 MIL JOINT USE AGREEMENT N	116 MIL LANDING RIGHTS Y
REPAIRS MAJOR MINOR	83 SEGMENTED CIRCLE N	117 BASED MIL & TYPE OPERS	118 TYPE & NO. MILITARY AIRCRAFT
AIRFRAME 102 Y 103 Y	93 HELIPAD ON AIRPORT N		
POWER PLANT 104 Y 105 Y	94 WEATHER STA ON ARPT N		
OXYGEN 106 HI PRESSURE N	95 ADMIN/TRML BLDG Y		
107 LO PRESSURE N	119 ESTD TOTAL ANNUAL MIL OPERS 0		BASED GEN AVIATION ACFT
OXY REPL BOTTLES 108 HI PRESSURE N	120 ESTD ANNUAL AIR TAXI OPERS 150		SINGLE ENGINE
109 LO PRESSURE N	ESTIMATED GENERAL AVIATION OPERATIONS		124 4 PLACE & OVER 16
110 AUX POWER FOR OPNL FAC	121 ANNUAL LOCAL 24411	122 ANNUAL ITINERANT 10462 / 123 TOTAL PEAK MONTH 4164	125 UNDER 4 PLACE 22
114 F-TYPE FUEL FOR PUBLIC SALE			126 MULTI-ENG 5
12 ☐ 18 ☐ 22 ☐ 30 ☐ *	134 FAR 139 CFR INDEX K		127 HELICOPTER 0
34 ☐ 35 ☐ 40 ☐ 45 ☐	145 UNICOM FREQ •122.8	147 TFC PATTERN ALT	128 SEAPLANE 0
SEAPLANE BASE DATA:	129 SHELTERED / 130 MOORING BUOYS	131 DOCK / 132 RAMP	133 BEACHING GEAR

137 REMARKS

E001 NORTH STUB TWY ONLY. GEN AVIATION OPN BASED ON M ASTER PLAN SURVEY.
E111 ESTAB PRIOR 1959
E114 FUEL CODES: 100LL, ~~[illegible]~~ 80
E145 UNICOM ATTENDED 0800-2100LCL.

135 LATEST INSP OR INFO REQ 08 NOV 1978 136 INSPN Y

NOTE: Except in #22 & 23, Y = Yes; N = No. "*" = See Remarks. Re #52, 1 = Approach; 2 = Approach and Roll Out; 3 = Approach, Mid-Point, and Roll Out. Re #45, L = Low, M = Medium, H = High.
Abbreviations are listed in FAA Contractions Handbook, Order 7340 ID

APPENDIX A (Page 7 of 8)

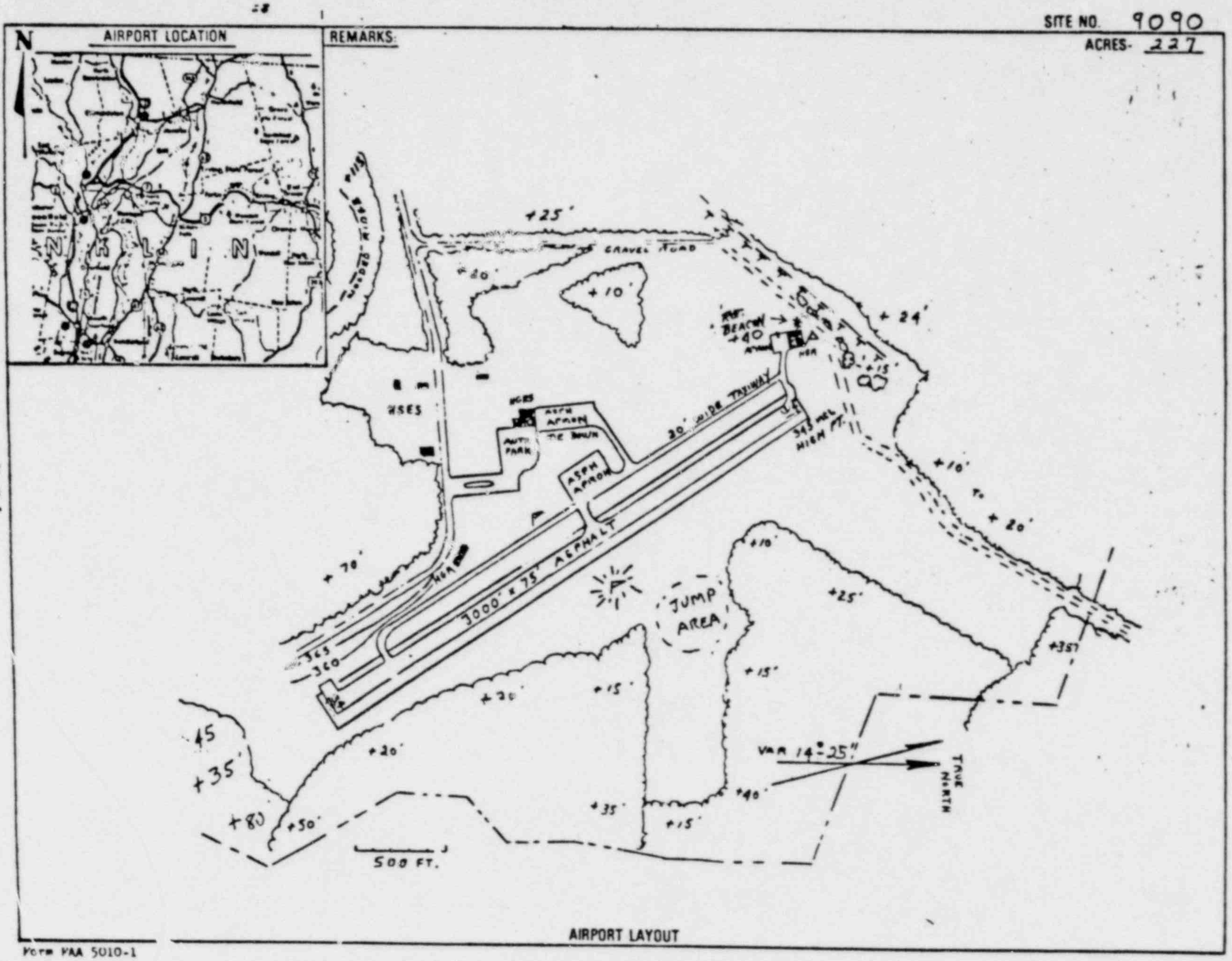
TURNERS FALLS AIRPORT

16 A

SITE NO. 9090

ACRES- 227

REMARKS:



Form FAA 5010-1

APPENDIX A (Page 8 of 8)
TURNERS FALLS AIRPORT

YANKEE NUCLEAR POWER STATION

Topic VI - 10.A: Testing of Reactor Trip System and Engineered Safety Features, Including Response Time Testing

Topic VI - 7.A.3: ECCS Actuation

I. Introduction

The goal of this assessment is to show that the Reactor Trip System and Engineered Safety Features test programs demonstrate a high degree of availability of the systems and that the response times assumed in the accident analysis are within the design specifications.

This assessment addresses Topics VI-10.A and VI-7.A.3 in one evaluation report since ECCS actuation is an integral part of the Engineered Safety Features System.

This report reviews the plant design to determine if all RTS and ESF components are included in the component and system tests, if the frequency and scope of the periodic testing is adequate, and if the test program meets the requirements of the following review criteria:

(a) General Design Criteria 21 - Protection system reliability and testability

(b) General Design Criteria 37 - Testing of emergency core cooling system

(c) Regulatory Guide 1.22 - Periodic testing of protection system actuation functions.

(d) Regulatory Guide 1.118 - Periodic testing of electric power and protection systems.

(e) Regulatory Guide 1.105 - Instrument setpoint

(f) Branch Technical Position ICSB 24 - Testing of Reactor Trip System and Engineered Safety Feature Actuation System Sensor response times.

(g) Branch Technical Position ICSB 25 - Guidance for Interpretation of General Design Criterion 37 for testing and operability of the ECCS as a whole.

(h) Standard Review Plan Section 7.2

In addition to those items required by the review criteria, the following will also be verified:

(a) That test conditions come as close as possible to the actual performance required by RTS and ESF.

(b) That compliance with the single failure criterion during testing is met.

(c) That the results of the licensee response time testing data for the RTS and ESF are within the delay times used in the FHSR accident analysis.

(d) That tests can be made to ensure the readiness or operability of system components.

(e) That the "Auto" mode of actuation does not inhibit the "Manual" mode of actuation, and vice-versa.

(f) That the power supplies satisfy the single-failure criteria.

(g) That the overlapping tests indeed overlap from one test segment to another.

(h) That equipment calibrations are adequate.

II. Testing of RTS and ESF at Yankee

1. Reactor Trip System - General Description

The Reactor Trip System automatically trips the reactor to protect against Reactor Coolant System damage and fuel rod cladding damage.

Two reactor trip breakers are provided to interrupt power to the control rod drive mechanisms, which are powered by Battery #2. These breakers are opened upon de-energizing the scram relays (which allow the breaker trip coils to operate) or by directly energizing the breaker trip coils.

The reactor shutdown function of the rods is completely independent of the normal control functions once power to the rods is interrupted. Response to any rod control signal is then impossible until the breakers are manually reclosed.

Power to the analog circuits is provided by the vital bus, which is fed by a dc-ac motor generator set. The dc motor is powered from station battery #1. Backup power to the vital bus is provided by manual action from a 480V station service bus section through a 480/120 volt transformer. Loss of vital bus power causes a reactor trip.

The scram breakers are tripped when the following trip actuation signals are received:

(a) Manual

A manual reactor trip allows the operators to trip the reactor. The manual actuation devices are independent of the automatic reactor trip circuitry and operate directly on the scram breaker trip coils.

(b) Main Steam Line Isolation Trip

This trip is generated upon occurrence of one or more of the following situations and protects the reactor against excessive voids and resultant high fuel temperature.

(1) High main coolant system pressure

(2) Low main coolant system pressure

(3) Low main steam line pressure

(4) Containment isolation system actuation (high vapor container pressure only)

(5) Manual initiation

(c) Low steam generator level

This trip is generated when a low level signal is received from any two of the steam generators (when above 15 MWe) and protects the reactor from a loss of heat sink.

(d) High Pressurizer Level

This trip is generated on a high water level signal from the pressurizer and is indicative of system overpressure.

(e) Low Main Coolant Flow (Steam Generator Δp)

This trip is generated on low differential pressure in any two of four steam generators (above 15 MWe) and is indicative of low main coolant flow.

(f) High intermediate range neutron flux start-up rate

This circuit trips the reactor when one of the two intermediate range channels reads above the trip setpoint (below 15 MWe). The intermediate range start-up range channels are separate from the intermediate range power range channels. This trip is generated to protect the reactor against transients caused by excessive start-up rates.

(g) High neutron flux level

This circuit trips the reactor when neutron flux levels within the intermediate range power range or power range exceed the trip setpoint. This trip protects the reactor against transients caused by excessive reaction rates.

(h) Turbine stop valve closure (Turbine Trip)

This trip occurs when one of the following signals has been received by the Turbine Stop Valve, and is designed to protect the turbine from being damaged:

(1) Any of the turbine protective trip conditions
(2) Manual Turbine Trip

(i) Low main coolant flow (MC pump current)

This trip is generated by a low or high current level in any 2 of 4 main coolant pump motors (above 15 MWe) and anticipates low main coolant flow.

2. Reactor Trip System - Testing

Provisions are made to manually place the output of the bistables or trip relays in a tripped condition for "at power" testing of all portions of each trip circuit, with the exception of the High Pressurizer Water Level, Turbine Trip, and Main Coolant Low Flow (SG Δp) trip units.

Provision is made for the insertion of a test signal in each channel. Verification of the test signal is made at specified points within the circuit in accordance with the appropriate surveillance procedure. This allows for testing and calibration of meters and bistables; transmitters and sensors are checked out against each other and against specified read-out equipment during normal power operation with the exception of the High Pressurizer Water Level, Turbine Trip, and Main Coolant Low Flow (SGΔp) trip units. In accordance with Technical Specifications, the High Pressurizer Water Level and Main Coolant Low Flow (SGΔp) functional tests are performed when shutdown longer than 24 hours if they have not been performed in the previous 31 days. The Turbine Trip functional test is performed prior to each start-up.

Each reactor trip system channel is checked by one or more of the following means:

(a) Varying the monitored variable

(b) Introducing and varying a substitute transmitter signal

(c) Cross-checking between identical channels or between channels which bear a known relationship to each other and have readouts available.

The means for manually bypassing channels or protective functions are administratively controlled, and access to all trip settings, module calibration settings, test points and signal injection points are controlled by keylock or administrative controls.

3. Engineered Safety Features - General Description

Engineered Safety Features are provided at the facility to mitigate the consequences of any design basis accident. ESF's have been designed to cope with any reactor coolant pipe break and with any steam or feedwater line break.

ESF's at Yankee are comprised of the following systems:

(a) Safety Injection System (ECCS)

(b) Recirculation System (ECCS)

(c) Containment Isolation System

A. Safety Injection System

Emergency core cooling is provided by the SIS which constitutes the Emergency Core Cooling System. The function of the SIS is to provide sufficient borated water to the reactor vessel to prevent core damage that would interfere with continued core cooling and to limit clad metal to water reactions to negligible amounts in the unlikely event of a major loss-of-coolant accident. In conjunction with the Recirculation System, the SIS provides for a constant and unobstructed flow of coolant to the reactor vessel.

SIS consists of the following:

(a) Three pumping trains, each composed of one low pressure safety injection pump and one high pressure safety injection pump.

(b) An accumulator filled with borated water.

(c) High pressure nitrogen storage flasks to pressurize the accumulator.

(d) A reserve of borated water in the safety injection tank.

A failure of the reactor coolant system boundary and its attendant loss of coolant results in a system depressurization and the initiation of a safety injection actuation signal (SIAS) as the pressure drops within the main coolant system or as the pressure increases within containment.

B. Recirculation System

The recirculation system is designed to provide long-term decay heat removal from the reactor core by recirculating cooled water collected in the vapor container sump back to the reactor vessel. The system is sized to provide sufficient flow to remove reactor decay heat. The vapor container functions as a passive decay heat exchanger transferring decay heat of the spilled water and vapor to the outside atmosphere through the steel shell of the vapor container. The system utilizes valves and piping, the LPSI and HPSI pumps, and other portions of the safety injection system.

A suction is taken from the vapor container through the sump strainer to either the low pressure or high pressure safety injection pumps. Recirculation flow returns to each loop of the reactor coolant system through the high pressure safety injection discharge piping.

Hot leg injection flow is from the high pressure safety injection discharge header and into the normal charging line which is connected to the hot leg of loop #4.

Flow elements are provided in each of the injection legs with readout in the control room to verify the continuation of flow. The pumps and valves are operated from the control room by the operator.

C. Containment Isolation System

Containment isolation is initiated automatically when a safety injection signal (non-essential valves only), a CIAS signal, or a manual signal is received by one of the two CIS logic trains.

Normally, open valves on each outgoing line used for the operation of the plant are closed upon actuation of CIS with the exception of those valves which perform safeguard functions. Incoming lines are isolated by two check valves on each line, one inside containment and one outside containment. The containment isolation actuation signal is provided by one of two identical pressure sensors. When either switch senses high containment pressure, the switch contact closes to energize a master trip relay which in turn energizes a pair of lockout relays which trip the isolation valves. No manual cutout is provided in the initiation circuit, and the integrity of each master trip relay coil is continuously monitored by an indication light next to each manual actuating switch. Provisions are made to manually bypass the actuation signal to each individual valve should the situation warrant such action.

4. Engineered Safety Features - Testing

A. The Safety Injection System is tested at each reactor refueling interval when main coolant pressure is less than 1000 psig. A pressure is applied to the proper side of the actuating pressure switches which initiates actuation of the system (with the exception of the accumulator). Once per 31 days when not shutdown, each pump, pressure switch, and active valve in the system is tested on a staggered test basis to ensure that it performs in accordance with technical specifications. This test program demonstrates operability of both the SIS and the Recirculation System.

Operability of the accumulator is based on:

(a) Verifying borated water level and concentration

(b) Verifying nitrogen supply bottle pressures

(c) Verifying isolation and relief valves position and/or operability

B. The Recirculation System valves and motors are tested in the same manner as those in the Safety Injection System as per Technical Specifications.

C. Containment Isolation System testing is performed during cold shutdown when the main coolant pressure is less than 300 psig. A pressure is applied to each sensor which initiates the system, and measurements are taken to ensure that each valve operates in accordance with Technical Specifications.

III. Evaluation and Conclusion

The Yankee test programs for the Reactor Trip System and Engineered Safety Features are in conformance with the review criteria listed in Section I of this assessment with the following exceptions:

(1) IEEE Std. 279 requires a capability for testing reactor protection system functions during reactor operation. The Turbine Trip, High Pressurizer Water Level, and Low Main Coolant Flow (SG Δ p) trip units are not testable during reactor operations. These trip units are normally tested in accordance with Technical Specifications (see Section II.2). It should be noted that the High Pressurizer Water Level and Low Main Coolant Flow (SG Δ p) trips are backup trips to the High Main Coolant System Pressure trip portion of the Main Steamline Isolation trip and the Low Main Coolant Flow (MC pump current) trip.

(2) Periodic tests for verification of system response times of reactor trip systems and engineered safety feature actuation systems are not performed at Yankee Rowe. Branch Technical Position EICSB 24 requires such periodic response time verification. Periodic tests for functional channel operability are performed in accordance with Technical Specifications. For those portions of trip systems that require a measured response time, response time tests are performed in accordance with Technical Specifications.

(3) Bypass and/or test annunciation is not provided for some of the RTS functions and ESF functions. IEEE Std. 279, section 4.13, requires continuous control room indication when any part of a protection system has been deliberately rendered inoperative. A system bypass or test is administratively covered in procedures and includes concurrence of the control room operator and Shift Supervisor.

(4) General Design Criteria 37 requires that testing of emergency core cooling systems adequately determine the proper functioning of the full operational sequence that brings the system into operation.

Yankee surveillance procedures require periodic testing of the individual components and subsystems of the Safety Injection System to assure proper functioning. These procedures, however, do not require the full nitrogen pressurization of the Safety Injection accumulator and the injection line leading to the loop injection point. These procedures adequately determine the proper functioning of the equipment required for the full operational sequence that brings the ECCS into operation.

YANKEE NUCLEAR POWER STATION

I. Topic XV-2: Spectrum of Steam System Piping Failures Inside and Outside of Containment (PWR)

I.A Introduction

A steamline break results in an increase in the flow of steam from one or more steam generators. The main steam system conducts steam from each of four steam generators to the turbine throttle valves. Steam leaves each of the steam generators through a 14-inch schedule 80 carbon steel pipe, which connects to a 24-inch main steam header. The flow is then directed via two 18-inch schedule 60 pipes to the turbine throttle valves.

Each of the main steamlines is protected by three safety valves with relief capacities and setpoints of 80,872 lb/hr at 935 psig, 118,260 lb/hr at 985 psig, and 573,329 lb/hr at 1035 psig. A non-return valve is located outside the vapor container in each steamline before the lines join to form the main steam header.

The non-return valves (NRV) were not originally designed for automatic closure. Thus, a break in any steamline that occurred on the turbine-side of the NRV's could result in blowdown from all four steam generators. During the Core XV refueling outage, however, each NRV was equipped with a stored energy actuator to provide for automatic closure within 3-5 seconds. In addition, each main steamline was equipped with three redundant pressure sensors to provide a signal to: (1) close all NRV's, (2) trip the reactor, and (3) isolate containment upon a 2 out of 3 coincidence of low pressure signals for any single main steamline. This design modification will prevent the blowdown of the remaining steam generators subsequent to automatic NRV closure and limit the primary plant cooldown transient to the equivalent of the inventory of a single steam generator. The same 2 out of 3 coincidence of low main steamline pressures will generate a newly-added permissive signal to trip the main condensate pumps. Thus, for steamline breaks inside containment, a high vapor container pressure signal will result in reactor trip, NRV closure, containment isolation, and main condensate pump trip. A description of this design modification was provided to the NRC in Attachment B of Reference I.1. In either case, an automatic trip of the main feedwater pumps occurs at power levels in excess of 15 MWe (Section I.B).

Prior to Core XV operation, safety analyses of the consequences of a main steamline break reflected the absence of automatic NRV closure. Typically, two separate types of steamline break analysis were performed: (1) a single steam generator blowdown inside the vapor container (break location upstream of NRV), (2) a four steam generator blowdown outside the vapor container (break location downstream of NRV). Reference I.2 (response to NRC IE Bulletin 80-04) discusses main steamline ruptures, focusing on the effects of feedwater system operation. Steam generator blowdowns both inside and outside the vapor container were considered; reactor plant performance and containment pressure response were also discussed. A subsequent letter to the NRC, Reference I.3, more specifically addressed the issue of environmental qualification of safety-related electrical equipment during a single steam generator blowdown inside the vapor container. Attachment D of Reference 1, the Core XV Performance Analysis Report, presents the analysis of a four steam generator blowdown to outside the vapor container. The objective of the analysis was to confirm that sufficient shutdown margin is required by the Technical Specifications to prevent the cooldown transient imposed on the reactor core following scram from causing a recurrence of criticality, which would raise operational concerns during the conduct of emergency recovery procedures.

The references cited above address the major considerations of a main steamline break, namely: (1) core thermal performance and radiological consequences of a break outside the vapor container, (2) shutdown margin adequacy for a four steam generator blowdown outside the vapor container, and (3) vapor container temperature and pressure response for a single steam generator blowdown inside containment. The analyses performed have characterized the steamline break accident in terms of the adequacy of plant design features and reactor protection system functions. These evaluations are discussed in the following paragraphs. The analysis of a four steam generator blowdown outside containment is only applicable for operation prior to the Core XV refueling outage, when automatic NRV closure was implemented. However, the consequences of this type of steamline break event are presented for information to show that recurrence of criticality is precluded without crediting the automatic NRV closure feature, which will prevent this cooldown transient during subsequent plant operations.

I.B Plant Response

A steamline break results in an increased rate of energy removal from the main coolant system. The amount of energy removed depends upon the mass of secondary coolant that escapes at the break location. Thus, the event is characterized in part by the number of steam generators that participate in creating the mismatch between increased energy removal via steam production and decreased energy production by the reactor core. The consequences of this mismatch are a main coolant system cooldown and depressurization and a secondary system inventory reduction and steam pressure decrease. If the break occurs inside the vapor container, containment pressure and temperature will increase. Numerous indications of this event are available to the operator, and numerous automatic protection system features are provided to shut down the reactor and ensure a safe recovery to a stable condition (see Sections I.C and I.D).

During the cooldown, total core reactivity is determined by several competing sources. The cold leg water temperature reduction adds positive reactivity via the moderator temperature coefficient. Insertion of control rods following scram and a safety injection flow of borated emergency supply water provide sources of negative reactivity. Core thermal performance prior to scram or safety injection is primarily affected by: (1) initial core power level or margin to subcriticality (operating-mode-dependent), (2) initial pressure and temperature of the primary and secondary coolants, (3) duration of the core power/steam flow mismatch prior to scram (e.g., amount of cooldown), (4) scram delay time associated with the trip-producing parameter, (5) power-dependent insertion-limit/scram worth, (6) moderator temperature coefficient, and (7) the break size. Core reactivity response following scram is determined by: (1) shutdown margin requirements, (2) initiation time of safety injection flow, (3) safety injection water temperature, flow rate, and boron concentration (4) steam generator feedwater initial flow rate, temperature, and post-trip operation. Finally, the vapor container pressure and temperature response for inside breaks the vapor container pressure and temperature response for inside breaks is affected by: (1) steam generator liquid and vapor mass and energy inventories, (2) feedwater initial flowrate, temperature, and post-trip operation, and (3) the high vapor container pressure trip-setpoint value that will cause NRV closure and reactor scram. The analyses to determine reactor plant and vapor container responses to a steamline break are performed separately, so that appropriate conservatisms may be freely applied to result in worst-case results.

Feedwater system performance affects the consequences of any steamline break. Two recent design modifications will reduce the severity of this transient and enhance the effectiveness of plant emergency procedures. They were discussed in Reference I.2 and include: (1) equipment modifications made during Core XIV operation to ensure that an automatic trip of boiler feed pumps occurs following reactor scram at power levels greater than 15 MWe (based upon turbine first stage nozzle pressure); and (2) installation of automatic trip logic for the condensate pumps, based on coincident signals for high vapor container pressure and low steamline pressure, performed during the Core XV refueling outage. These changes ensure that feedwater addition will be rapidly terminated subsequent to reactor scram to minimize reactor cooldown effects and, for rupture inside containment, to minimize the vapor container pressure and temperature increases. In addition, emergency procedures emphasize isolation of feedwater flow to the affected steam generator and selective feedwater addition to steam generators that are not losing their inventories out the break.

I.C Plant Protection

A reactor trip during a steamline break accident may result from the following trip conditions:

(1) high neutron flux trip,

(2) low steam generator water level (>15 MWe power level),

(3) low main coolant system pressure,

(4) high vapor container pressure, or

(5) low main steam system isolation trip logic (effective Core XV).

I.D Event Indications

Indications of a steamline break include the following:

(1) reduced steam generator pressure,

(2) reduced main coolant system pressure and temperature,

(3) abnormal steam generator level and steam flowrate indications,

(4) high containment pressure,

(5) mismatch in steam/feedwater flowrates,

(6) actuation of safety injection,

(7) containment isolation actuation,

(8) automatic closure of all NRV's (effective Core XV).

I.E Analysis

I.E.1 General Information

The maximum credible size for a main steamline break results from a circumferential rupture of the 24-inch main steam header. Following the Core XV refueling outage, the automatic closure of all NRV's will prevent the blowdown of all steam generators. Analysis has been performed, however, for the four steam generator blowdown. It represents the most severe case, since all four steam generators could suffer complete losses of inventory until atmospheric pressure is reached. Thus, this break location results in the largest cooldown transient and the largest addition of positive reactivity. With respect to (1) concerns of acceptable core thermal performance (see Section I.E.2) and (2) confirmation of adequate shutdown margin (see Section I.E.4), the four steam generator blowdown is a limiting event if credit is not taken for automatic closure of the NRV's. The core thermal performance is based upon analysis presented in Reference I.4, the Core XI refueling submittal to the NRC. The most recent confirmation that recurrence of criticality is precluded is contained in Reference I.1, the Core XV refueling submittal.

A rupture that occurs inside the vapor container must be located upstream of the NRV's. These valves were originally designed to act as reverse-flow check valves. Thus, the analysis for a steamline break inside containment applies to a single steam generator blowdown and considers all possible sources of mass and energy (see Section I.E.3). This analysis was presented in Reference I.3 for purposes of environmental qualification of safety-related electrical equipment.

I.E.2 Analysis: Core Thermal Performance Following a Main Steamline Header Break Outside Containment

Two initial conditions were assumed for a four steam generator blowdown to outside the vapor container: full load and no load. The following additional assumptions apply to predict a conservative DNB ratio:

(1) the moderator temperature coefficient was most-negative and varied with temperature (uncertainties included);

(2) minimal full temperature reactivity feedback (uncertainties included);

(3) minimum available scram rod worths minus allowance for most reactive rod failing to insert (6.93%$\Delta\rho$-full load/4.72%$\Delta\rho$-no load, consistent with technical specifications);

(4) steam-phase blowdown assumed to maximize energy removal;

(5) feedwater flow decreases to zero at 30 seconds following trip on nuclear overpower signal; and

(6) boron concentration of safety injection water initially at 2200 ppm and boron reactivity worth of one percent $\Delta\rho$ per 147 ppm assumed.

The plant transient response was calculated using the GEMINI-II computer code; the hot channel DNBR analysis was performed using the COBRA-III-C computer code.

The break initiated at full power results in a high neutron flux trip signal at approximately 4 seconds; the low-initial power case results in this trip at 28 seconds after event initiation. Safety injection is initiated for the full- and zero-initial power cases at 9 and 10 seconds, respectively, based upon a low main coolant pressure signal. The respective peak power levels were determined to be 125 percent and 113 percent of rated power, respectively.

A DNB analysis was made for these two cases using the Westinghouse W-3 correlation in the COBRA-III-C subchannel analysis code. In both cases, the minimum DNBR exceeds 1.30 and no fuel melting is predicted.

I.E.3 Analysis: Vapor Container Pressure Response Following a Main Steamline Break Inside Containment

Each of the four main steamlines contains an NRV located outside the vapor container. These valves act as reverse-flow check valves and are being provided with automatic actuators during the Core XV refueling outage to permit automatic and rapid closure upon receipt of either a low steamline pressure or a high vapor container pressure signal. A steamline break that occurs inside the vapor container could result in a discharge inside containment of the contents of a single steam generator and the associated main steam piping upstream of the NRV's of the remaining steam generators.

Reference I.3, provided to the NRC, described the consequences of this event with respect to vapor containment pressure and temperature response. The analysis was performed using the RELAP4 computer code for blowdown calculations and the CONTEMPT-LT026 code for the containment thermodynamic response.

The most adverse initial conditions were assumed for the analysis in Reference I.3, covering the full range of operating conditions from zero to full power operation. The initial power level was assumed to be 112% of 618 MWt, the maximum high neutron flux trip point, and was held constant during the calculation until a reactor trip occurred based on high vapor container pressure. A steam generator inventory of 31500 lbm was assumed for conservatism, although the inventory at full power operation is actually 20000 lbm. Full feedwater system operation with all feed pumps was assumed, with all feedwater being directed to the steam generator blowing down through the ruptured steamline. Thus, the feedwater control valve of the affected steam generator was assumed to open fully following the break. The feedwater system operation was reviewed in the Reference I.2 response to IE Bulletin 80-04. A double-ended guillotine break of the main steamline was assumed, to maximize the blowdown rate and energy addition to containment.

In addition to the assumptions discussed in the preceding paragraph, other initial conditions and assumptions are listed below:

1. Feedwater System Operational Mode - three boiler feed pumps operational,
2. Break Size - double-ended guillotine (.8522 ft^2),
3. RCS Main Coolant Pumps - remain active,
4. Primary Loop Heat Transfer to SG Liquid - variable heat transfer coefficients used, but degradation in heat transfer due to reduced area during bundle uncovering was neglected,

5. SG Tube Area - the entire SG tube area was assumed active,
6. Reactor Trip Signal - occurs 1.0 second after vapor container high pressure signal,
7. Feedwater Termination - occurs 10.0 seconds after reactor trip signal,
8. Flow to Turbine - zero at all times.

The RELAP4 blowdown analysis indicates a trip on high containment pressure at 5 seconds, but for conservatism the reactor trip was delayed until 11 seconds. This trip time includes a 1 second delay time. The affected steam generator was emptied via blowdown at 112 seconds into the transient

A CONTEMPT code thermodynamic analysis was then performed. The maximum containment pressure was determined to be 31.7 psig, which is less than the containment design pressure of 34.5 psig. This result was transmitted together with temperature response to the NRC per Reference I.3 to confirm vapor container integrity using conservative assumptions for a major steamline break.

On May 23, 1980, the NRC issued Memorandum and Order CLI-80-21 which required that the DOR Guidelines and NUREG-0588 form the requirements that must be met regarding environmental qualification of safety-related electrical equipment. In compliance with this order, Reference I.5 presented results of an exhaustive re-evaluation of environmental qualifications for safety-related electrical equipment at Yankee Rowe required to function under the harsh environments associated with design basis accidents.

Steamline breaks inside and outside the vapor container were reviewed in confirming the availability of electrical equipment and instrumentation required to bring the plant to a cold shutdown condition.

I.E.4 Analysis: Primary System Cooldown Response and Core Reactivity Following a Main Steamline Break

The subject of core reactivity response during the cooldown imposed after a major steamline break was discussed in the Reference I.2 response to IE Bulletin 80-04. A review was performed of the potential for recurrence of criticality following scram, which could result if core shutdown margin was reduced to zero during cooldown. In Reference I.2, two design modifications were discussed that reduce the amount of cooldown during a steamline break. These modifications, discussed in Section I.A, include automatic tripping of the main condensate pumps on coincident signals of high containment pressure and low steamline pressure, and ensured main feed pump auto-trip at power levels greater than 15 MWe. These design changes were implemented in addition to the automation of non-return valves, which provides for main steam system isolation upon receipt of either a high containment pressure or low steamline pressure trip signal.

The potential for a return to power exists when the primary coolant system is rapidly cooled by increased energy removal due to the blowdown of secondary coolant. The combined effect of all various reactivity contributions determines whether recriticality can occur following shutdown. Ultimately,the amount of cooldown depends upon the available secondary coolant inventory and feedwater system performance following reactor trip. Following automation of NRV closure during the Core XV refueling outage, blowdown of more than one steam generator either inside or outside containment will be prevented when credit is taken for either (1) automatic closure of the NRV's in fulfilling their function as isolation valves, or (2) prevention of reverse flow through the NRV's of the remaining unbroken steamlines in fulfilling their function as reverse flow check valves.

Analysis was performed for each reload core to determine the shutdown margin required to prevent recurrent criticality due to a cooldown associated with complete blowdown of all secondary coolant from each steam generator. Most recently, the Core XV analysis presented in Reference I.1 indicated that recriticality was prevented without crediting main steam system isolation on reactor trip on low main steamline pressure. The Core XV analysis is similar to the Core XIV analysis. The Core XIV analysis demonstrated for both full power and zero power cases that if cooldown of the primary loop to 70°F is assumed following shutdown, the core retains sufficient shutdown margin to ensure subcriticality. This calculation included moderator temperature defect, fuel temperature Doppler defect, provision for the most reactive control rod stuck out, boron insertion with safety injection, and conservative nuclear design uncertainties. Based upon performing this calculation for Core XV, the required shutdown margin for Mode 3 operation at hot subcritical conditions was increased from 4.72%$\Delta\rho$ to 5.5%$\Delta\rho$ to provide additional margin to subcriticality. The value is reduced to 4.72% during plant cooldown as shutdown requirements lessen. Thus, rod insertion limits are imposed to provide adequate shutdown margin to preclude recriticality, even when main steam isolation is not credited and blowdown of all four steam generators is assumed.

An evaluation was performed by the Westinghouse Corporation of reactor vessel integrity during a steamline break cooldown transient. This analysis shows that acceptable vessel integrity for the primary cooldown resulting from blowdown of the entire inventory of a single steam generator. The minimum expected temperature for this event is 300°F, if the event is initiated at a maximum allowable value of T_c = 519°F (Technical Specification value of ≤ 515°F plus 4°F uncertainty). Westinghouse analysis showed that system repressurization during the cooldown was permissible to values in excess of the pressurizer code safety setpoint of 2500 psig. In observance of vessel integrity consideration, plant emergency procedures for loss of secondary coolant currently instruct the operator to terminate safety injection when the cold leg temperature of any loop reaches 310°F, but only after verifying the existence of adequate subcooling and system pressure control. However, operator termination of SI flow is not required to prevent violation of vessel integrity limitations during cooldown.

I.F References

I.1 FYR 81-52, YAEC Letter to NRC, Core XV Refueling Proposed Change No. 173, 26 March 1981.

I.2 YAEC Letter WYR 80-50 to NRC, Response to IE Bulletin No. 80-04, 8 May 1980.

I.3 YAEC Letter WYR 80-62 to NRC, Environmental Qualification of Electrical Equipment, 5 June 1980.

I.4 YAEC Letter to NRC, Proposed Change No. 115 (Core XI Refueling), 29 March 1974.

I.5 YAEC Report 1227, Environmental Qualification of Safety-Related Electrical Equipment, October 1980.

II. Topic XV-3: Loss of External Load (Turbine Trip), Loss of Condenser Vacuum, Steam Pressure Regulator Failure

II.A Introduction

A rapid and large reduction in power demand on the reactor while it is operating at full power results in a corresponding reduction in the rate of heat removal from the main coolant system. Such an incident could lead to system overpressurization if suitable protection was not provided.

The most probable cause of a rapid loss of load is a turbine trip. Following a turbine trip, the reactor would be tripped directly (except for power levels below 15 MWe) from a signal derived from the turbine stop valves. The steam bypass system would accommodate the excess steam generation. The steam bypass functions to limit the increase in main coolant temperature and pressure for this transient.

Primarily, the turbine trip event is analyzed to evaluate the protection provided to prevent overpressurizing the primary system. The following design features exist to prevent primary system overpressurization:

1. Pressurizer power operated relief valve,
2. Steam dump system (steam bypass valve),
3. Pressurizer spray,
4. Main coolant loop safety valves,
5. Pressurizer safety valves,

6. Steam generator safety valves,

7. Charging volume and control system, and

8. Reactor protection system (reactor trip).

The reactor protection system would scram the reactor, in the event of a turbine trip, on a variety of signals listed below:

1. Turbine trip signal,

2. High pressurizer level trip,

3. Steam generator low level trip,

4. High main coolant system pressure (implemented during Core XV refueling outage).

II.C Plant Response

Both the pressurizer power-operated relief valve and the steam dump valve are provided to prevent the spring-loaded safety valves from opening and are not intended to be part of the system overpressurization protection. In the event the steam dump valves fail to open following complete loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer water level signal. The steam generator and pressurizer safety valves are sized to protect the main coolant system and steam generators against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power-operated relief valve, automatic rod control, or direct reactor trip following turbine trip.

The steam generator safety valve capacity is sized to remove the maximum calculated steam flow from the steam generator (105 percent of maximum guaranteed steam flow) without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is based on a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load, assuming operation of the steam generator safety valves. The pressurizer safety valves are then able to maintain the main coolant system pressure within 110 percent of design pressure without direct or immediate reactor trip action.

In order to demonstrate that the main coolant system is adequately protected from overpressurizing during a complete loss of load transient, the analysis did not take credit for either the steam dump system or the pressurizer power-operated relief valve. When credit is not taken for the immediate turbine trip or subsequent steam generator low level signal, the reactor is tripped by the high pressurizer level trip (or the high main coolant loop pressure logic, effective with Core XV operation).

Loss of condenser vacuum can also result in turbine trip and precludes use of the steam dump system. Since the steam dump is assumed to be unavailable in the analysis of the turbine trip event, no additional analysis is necessary to include separate effects from loss of condenser vacuum. Also, the plant design does not include a steam pressure regulator so the failure of this device is not an applicable design basis event.

II.D Analysis

The analysis of the complete loss of load, reported in Reference II.4 for Core XI, was performed using the GEMINI-II digital computer program to simulate the nuclear steam supply system response. The hot channel analysis was performed with the COBRA-III-C computer program.

The following assumptions were made:

1. A turbine trip occurs instantaneously without activating a reactor trip;
2. The moderator coefficient is the least negative expected at beginning of life for power operation, including calculational uncertainties;
3. The beginning of life fuel temperature coefficient is increased by 25 percent to account for calculational uncertainties;
4. The pressurizer control is assumed to be in the manual mode. Thus, no credit is taken for the effect of the spray or letdown systems in reducing the rate of pressure increase;
5. One safety valve in each steam generator is inoperative.

Results show that high level trip occurs at 20 seconds after initiation of the incident. The pressurizer safety valves act to keep the main coolant system pressure below 2545 psia and the operational steam generator safety valves limit secondary pressure to 1040 psia. The minimum DNB ratio and fuel temperatures improve during the transient.

A bounding analysis was performed for the reference Core XI analysis using the design value of moderator temperature coefficient at BOC. In general, this incident is not sensitive to minor changes in core parameters. In addition to the Core XI reference analysis, numerous parametric analyses were performed in support of the Core XIV reload submittal. These parametric studies were requested by NRC (Reference II.1), submitted via Reference II.2, and subsequently approved via issuance of Amendment No. 54 (Reference II.3) to the Facility Operating License. These sensitivity studies on moderator temperature coefficient and Doppler coefficient demonstrated the minor impact of core physics parameters on the loss of load transient. These parametric studies provided in Reference II.2 conservatively bound Core XV core parameters, reported in Reference II.5. In addition, the high main coolant pressure trip function implemented during the Core XV refueling outage will reduce challenges to code safety valves for this event.

II.E References

II.1 NRC Letter from D.L. Ziemann to R.H. Groce, YAEC, 8 November 1978.

II.2 WYR 78-99, YAEC Letter from D.E. Vandenburgh to NRC, Additional Information -- Core XIV Refueling, 21 November 1978.

II.3 NRC Letter from D.L. Ziemann to R.H. Groce, YAEC, Amendment No. 54 (Core XIV), 6 December 1978.

II.4 YAEC Letter to NRC, Proposed Change No. 115 (Core XI Refueling), 29 March 1974.

II.5 FYR 81-52, YAEC Letter to NRC, Core XV Refueling Proposed Change No. 173, 26 March 1981.

III. Topic XV-5: Loss of Feedwater Flow

III.A Introduction

The feedwater system is designed to provide a continuous flow of water to the four steam generators during normal plant operation. A rapid and large decrease in feedwater flow, when operating at power without a corresponding reduction in steam flow, would lead to a decrease in water inventory of the steam generators. The main feedwater system consists of three electric motor-driven parallel boiler feedwater pumps with a common suction and discharge header that provide normal feedwater flow. The pumps take suction from the condenser via three parallel condensate pumps. Flow is controlled by four separate control valves. In the event of a total loss of main feedwater, an emergency feedwater system is available to provide water to the steam generators. Furthermore, the turbine trip follows reactor trip and the steam dump system is available for decay heat and stored energy removal.

The emergency feedwater system includes one positive-displacement steam-driven pump with minimum capacity of 80 gpm. This pump does not rely on electrical power and can function following total loss of AC. Steam to drive the pump may be supplied from either the main steam system or the auxiliary oil-fired boilers. In the event of total loss of AC, natural circulation cooling would be procedurally controlled and this steam-driven pump would provide feedwater to the steam generators. Decay heat removal would occur via blowdown through the steam generator safety valves. The steam-driven feed pump takes steam from upstream of the non-return valves and suction from the demineralized water storage tank. A back-up system to supply water to the steam generators in the event of failures in the emergency feedwater system is the plant's three charging pumps with a total capacity of approximately 100 gpm (33 gpm/pump). The system is connected permanently by a spool piece that connects to the main feed line header. The charging pumps can take suction flow from the 135,000 gallon Primary Water Storage Tank. High Pressure Safety Injection and Low Pressure Safety Injection pumps provide another back-up source to supply water through the same permanently connected spool piece used for the charging pump path. More than 100 gpm flow is available from the combination of a single high pressure safety injection (HPSI) pump and low pressure safety injection (LPSI) pump, and there is a total of three HPSI pumps and three LPSI pumps. Upon loss of AC power, and LPSI pumps can be directly powered by the diesel generators.

In addition to the steam-driven pump, a modification to increase the emergency feedwater capabilities is being made during the Core XV refueling outage. This modification was described to the NRC in Reference III.1, Attachment C. Two motor-driven emergency feedwater pumps of >150 gpm capacity each will be installed to increase the capability and redundancy of the emergency feedwater system, thus providing a more reliable method of mitigating the consequences of a complete loss of main feedwater. Additional piping will be installed to permit feedwater addition, with either pump, through either the normal feedwater piping via the steam-driven emergency pump header or through the blowdown piping via the alternate emergency feedwater header. These motor-driven pumps can be started either remotely from the control room or locally by operator action. Upon loss of AC, one of the three on-site diesel generators can be used to power either of these motor-driven pumps. Remote indication of emergency feedwater flowrates through the normal feedwater path will be available in the control room. Local flow indication will be available in the combined pump minimum-recirculation-flow piping.

Thus, a redundant and versatile emergency feedwater system is available, consisting of one >80 gpm steam-driven pump and two >150 gpm motor-driven pumps. Also, the 100 gpm total charging pump capacity and the safety injection pumps (one HPSI and one LPSI provide >100 gpm) are available as back-up feedwater sources. Adequate emergency feedwater supplies are provided from these numerous alternative sources of pumping capacity.

III.B Plant Response

The feedwater system is designed to minimize the probability of complete loss of flow to the steam generators. Three boiler feed pumps having a common suction and discharge are provided and the feedwater control valves are designed to fail in-position on loss of air from the air controller system. However, in addition to loss of power to the main feedwater pumps, a complete loss of normal feedwater flow might occur from:

1. a malfunction in the feedwater regulating system that drives all the feedwater regulating valves closed;

2. a rupture located downstream of the feedwater check valves, such as at the feedwater header; or

3. an operator error that leads to closure of all feedwater regulating valves when feedwater control is in manual mode.

Feedwater flow to the steam generators would not stop instantly for any of these events, but would begin rapid coastdown when reactor trip occurs. Primary system pressure increases with the reduction in steam generator inventory. Pressurizer safety valves and main coolant loop safety valves are available to limit the pressure increase while core power reduction terminates the heat addition; however, challenges to these valves are not expected to occur.

III.C Plant Protection

Protection system diversity exists to provide a reactor trip signal based upon either low steam generator level, high pressurizer water level, or high main coolant system pressure (implemented during Core XV refueling outage).

III.D Analysis

Reference III.2 provided to the NRC an analysis of a total loss of feedwater event from a full power operating conditions (increased 3% to 618 MWt to include uncertainties). A step-change of the normal feedwater flowrate to zero was assumed. The steam bypass system was conservatively assumed to function normally after turbine trip to maximize both fluid inventory loss and energy removal. No credit was assumed for pressurizer spray effects or for letdown and charging system operation to reduce the primary pressure increase. Conservative values of the moderator temperature and fuel temperature coefficients of reactivity were assumed, in addition to assuming the Technical Specification minimum main coolant flow and maxi- steam pressure during normal operation. The maximum allowed core inlet temperature and nominal main coolant pressure were assumed.

No credit was taken for emergency feedwater from the >150 gpm motor-driven feedwater pumps being installed during the Core XV refueling outage. Emergency feedwater from the steam-driven pump at 80 gpm was delayed for fifteen minutes. The two main coolant pumps normally powered from the turbine generator were assumed to trip after sixty seconds. The pumps were assumed to be restarted within ten minutes to provide additional heating of the primary system.

Analysis of the loss of feedwater flow transient was performed using two basic methods: 1) the GEMINI-II computer code, and 2) calculations which considered the total conservation of mass and energy of both the primary system and the secondary system.

The GEMINI-II computer code was used to determine the initial transient response of the primary and secondary system and to provide a comparison to the previous analysis, Core XI, which was submitted to the NRC via Reference III.2. Primary system structural heat capacity is modeled with GEMINI-II. Additional calculations considered the total mass and energy of the primary and secondary systems and supplemented the GEMINI analyses by accounting for the following parameters not modeled in GEMINI-II:

1. main coolant pump energy input to primary system,

2. steam generator secondary side structural heat capacity, and

3. ANS 5.1 decay heat (Reference III.3).

A low steam generator level trip occurs at 18 seconds. Results showed that primary pressure peaked at 2178 psia at 22 seconds, followed by rapid decrease to 1860 psia at 60 seconds. The maximum pressure is significantly lower than the pressurizer power operated relief valve setpoint of 2400 psia. The turbine throttle valve closed at 20 seconds and steam pressure settled to the steam bypass system setpoint of 760 psig for the duration of the transient. Primary system temperature reached a maximum of 533°F, but settled to the temperature that corresponds to the steam bypass system setpoint, approximately 514°F. This is the plant no-load T_{ave} value.

Steam generator inventory loss was maximized by not assuming a reduced rate of heat transfer with decreasing water level and by not crediting the availability of the motor-driven emergency feedwater pumps. Without accounting for heat transfer rate degradation, the minimum inventory was 14 percent of the initial value at 74 minutes, when inventory recovery occurred. This result is made more conservative since no credit was taken for heat losses to ambient from either the primary or secondary system. Neglecting feedwater addition from any auxiliary source, steam generator inventories are sufficient to delay dryout for approximately 40 to 70 minutes when AC power is available, depending upon initial conditions.

If AC power was not available, the main coolant pumps and steam bypass system would not be in operation. Neglecting main coolant pump heat addition and limiting the steam generator blowdown to the secondary code safety valves results in more favorable steam generator inventory retention. Dryout times for this case are approximately 85 minutes. Thus, AC power was assumed to be available for conservatism.

The most recent analysis of this event used Core XV nuclear design parameters and demonstrated the conservatism of the Reference III.2 analysis, which utilized Core XIV values. In conclusion, the combination of the reactor protection system and the auxiliary feedwater system ensures the integrity of the core, and primary and secondary system pressure boundaries by (1) reactor trip on low steam generator water level, and (2) auxiliary feedwater flow sufficient to assure adequate steam generator liquid inventory for primary system cooldown, decay heat removal, and main coolant pump heat removal for the entire course of the event. The addition of (1) the high main coolant loop pressure instrumentation and trip logic, and (2) the two >150 gpm motor-driven auxiliary feed pumps (which may be operated from the control room) during the Core XV refueling outage provides even greater protection for this event.

References

III.1 FYR 81-52, YAEC Letter to NRC, Core XV Refueling, 26 March 1981.

III.2 WYR 79-104, YAEC Letter to NRC, Yankee Rowe Loss of Feedwater Analysis, 12 September 1979.

III.3 ANS Standard 5.1, Decay Heat Power in Light Water Reactors, September 1978.

IV. Topic XV-7: Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

IV.A Introduction

The main coolant pumps do not possess a large inertial mass because they were not designed with flywheels. Thus, following loss of power to a pump's motor the flow rate rapidly coasts down. Tests were conducted in 1971 to determine the pump coastdown characteristics. Results showed that the flowrate from a single pump, which was tripped from normal T_{ave} and pressure conditions, was reduced by one-half in 0.58 seconds. Loss of pumping power from all four main coolant pumps is prevented by using independent sources of power for pump pairs: one pump is powered by an incoming line and another is powered by an independent incoming powerline; the other pair is powered from a transformer connected to the main generator leads. The loss of two main coolant pumps during four-loop operation is considered to be the most severe anticipated operational occurrence. Reactor protection for this event is provided by either a low main coolant flow trip or a trip on high or low current to (1) two or more pumps during four-loop operation or, (2) one or more pumps during three-loop operation. Currently, three-loop operation is not permitted.

Loss of flow from a single main coolant pump during four-loop operation, will not result in direct reactor trip based upon a high or low pump current signal. Loss of power to all four main coolant pumps with reactor trip, however, is a more severe event than rotor seizure or shaft inertial mass of the pumps. This was confirmed in Reference IV.1, the performance analysis for Core XI. The Core XI analysis showed that the steady-state DNB ratio during full power four-loop operation at 618 MWt (includes 3% uncertainty), following instantaneous loss of flow from a single pump without reactor trip, is greater than the minimum DNB ratio following complete loss of pumps, reactor trip, and flow coastdown during four-loop operation at 618 MWt. The analysis of the complete loss of flow accident is discussed in the next section.

IV.B Analysis

The results of the complete loss of flow accident performed for Core XI indicated that the DNBR ratio decreased to below 1.30 in 2.16 seconds after pump coastdown was initiated. No credit was assumed for the coastdown delay associated with the two pumps powered by the turbine-generator. Using conservative heat transfer correlations, the maximum clad temperatures remained below 1200oF. Assuming that all cladding in excess of 1100oF fails, the amount of failed fuel was estimated at less than 1.25 percent. This analysis was approved by the NRC in Reference IV.2. The minimum DNB ratio for three-loop steady-state operation at full rated power was 2.54 (normally, power level during approved three-loop operation is restricted to 75% rated). A seized-rotor or shaft-break event could not result in a more limiting DNB ratio. A two-pump loss of the flow and reactor trip during normal three-loop operation resulted in a minimum DNB ratio of 2.51 seconds at 1.6 seconds. Thus, seized-rotor or shaft-break events that result in loss of a single pump without reactor trip are less limiting than a complete loss of pumps with reactor trip, and the DNB ratio for these events is greater than 1.30.

A more recent analysis of the four-loop loss of pumping power accident was performed for Core XV and submitted to the NRC via Reference IV.3. This analysis was performed using methods similar to those approved by Reference IV.2 for the Core XI analysis. Namely, the CHIC-KIN code was used to determine core power and fuel pin thermal responses and the COBRA-III-C subchannel analysis code was used to determine the hot channel thermal-hydraulic response and minimum DNB ratio. Results for Core XV presented in Reference IV.3 indicated that no fuel damage was expected following a complete loss of pumping power during four-loop operation. Based upon the comparison made for Core XI, a seized-rotor or shaft-break is a less limiting event.

The seized-rotor and shaft-break events would not result in DNB ratios much less than 2.54 or 2.51, and are not the limiting loss of flow event for the plant. No fuel damage is predicted; no radiological consequences occur. If this event occurs in coincidence with a turbine trip and loss of offsite AC power, the core would be cooled by natural circulation using the remaining three loops. Sufficient secondary coolant inventories exist for adequate primary-to-secondary heat transfer and controlled cooldown.

IV. References

IV.1 YAEC Letter to NRC, Proposed Change No. 115 (Core XI Refueling), 29 March 1974.

IV.2 DOL/AEC Letter to YAEC, Amendment No. 9, 30 July 1974.

IV.3 YAEC Letter FYR 81-52 to NRC, Core XV Refueling Proposed Change No. 173, 26 March 1981.

V. Topic XV-9: Startup of an Inactive Loop (PWR)

V.A Introduction

If a reactor coolant loop is isolated from the remainder of the primary system and subsequently opened into the system, without first matching the boron concentration and temperature of the isolated loop to the system, an increase in core reactivity and power may occur. Such an incident could lead to loss of margin to core thermal limits unless suitable procedures and protection are provided.

Operation of the plant with only three active loops is currently not permitted. However, when supporting LOCA analysis for this mode is available, three-loop operation at power levels up to 75 percent is anticipated. Normal plant operation presently requires closed and locked isolation valves with power removed, in both the hot and cold legs of the inactive loop on the primary side and in the main steamline on the secondary side of the inactive loop. The Technical Specifications require a subcritical condition of at least a 1% $\Delta\rho$ before an isolated loop may be placed into service.

When the plant is operating with one loop isolated from the remainder of the primary system, the isolated loop is allowed to cool below the temperature in the active loops. In order to bring the isolated loop back into the system, administrative procedures have been established to prevent a mismatch in either boron concentrations or coolant temperatures between the isolated and active loops. These procedures require the isolated loop temperature to be greater than or equal the highest cold leg temperature of the operating loops. The boron concentration of the inactive loop must be greater than that of the active portion of the primary system prior to opening the loop isolation valve and placing the loop into service.

In order to provide additional protection against the occurrence of a cold water incident, interlocks have been placed on the cold leg isolation valve controls so that they cannot be opened unless the isolated loop temperature is within 30°F of the hottest active loop cold leg temperature. Administratively, this temperature difference is restricted to 20°F. Furthermore, the slow-opening loop isolation valve permits gradual mixing of the isolated loop water with water of the active loops. The loop transit time is short compared to valve opening time, which favorably increases mixing.

Administrative procedures for handling the startup of an isolated loop require at each stage of the startup that the boron concentration in the inactive loop is always greater than or equal to the concentration in the remainder of the primary system. In order for a lower boron concentration to exist in the isolated loop when brought into service, more than two administrative procedures will have to be violated.

Although the occurrence of the cold water incident is considered highly unlikely, an analysis has been performed to demonstrate that the reactor protective system precludes fuel damage for the largest temperature mismatch possible between the active and isolated loops. This analysis was submitted via Reference V.1 and approved in Reference V.2.

V.B Analysis

The following assumptions were used in the analysis:

1. The temperature of the borated water in the isolated loop is 68°F, the lowest value considered possible for this incident;
2. The moderator temperature coefficient is the most negative value expected;
3. The fuel temperature coefficient is the least negative value expected at hot, operating conditions, including calculational uncertainties. This will result in the largest power overshoot above the high neutron flux trip setpoint;
4. The initial core flow rate is 78 percent of the normal four loop flow rate. The flow increase resulting from the isolation valve opening is proportional to the cross-sectional flow area of the gate valve;
5. The core inlet temperature variation with time is determined from a complete mixing model.

Results show that core inlet temperature and primary pressure decrease with time as the cold water from the isolated loop mixes with the higher temperature water in the reactor inlet plenum. Decreasing temperature adds reactivity to the core, reactor power rises, and the high neutron flux trip point is reached at 9 seconds. The minimum DNB ratio during the transient is greater than 2.97 and peak fuel temperature does not exceed 3485°F.

More recently, this event was analyzed using Core XV nuclear design parameters. Results were submitted to the NRC in Attachment D to Reference V.3. Since the Core XV moderator temperature coefficient of reactivity is less negative than assumed for Core XI, results for Core XI are bounding.

In conclusion, adequate margin to DNB exists for this accident even for the highest possible temperature mismatch between the active loops and the isolated loop. In addition, procedural controls and the cold leg isolation valve interlock minimize the probability that the event will occur. The high neutron flux trip provides automatic protection.

V.C References

V.1 YAEC Letter to NRC, Proposed Change No. 115 (Core XI Refueling), 29 March 1974.

V.2 DOL/AEC Letter to YAEC, Amendment No. 9, 30 July 1974.

V.3 FYR 81-52, YAEC Letter to NRC, Core XV Refueling, 26 March 1981.

YANKEE NUCLEAR POWER STATION

VI. Topic XV-10: Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant

VI.A Introduction

Boron concentration in the main coolant system is diluted during normal operations to compensate for fuel depletion and changes in xenon inventory. Strict administrative controls are exercised to limit both the rate and amount of any change in boron concentration required for reactivity control.

Boron dilution is accomplished according to procedure by transferring charging pump suction from the Low Pressure Surge Tank to the demineralized water supply. Technical Specifications prohibit dilution during any mode of operation unless at least one reactor coolant pump or the shutdown cooling system is in operation. The reactivity change must be $\leq 1.5(10)^{-4} \Delta\rho$ /sec with main coolant temperature $\geq 250^{o}$F.

The demineralized water is introduced into the primary coolant using the charging and volume control system. Normally, charging pump flow passes through the shell side of the feed and bleed heat exchangers for warming before entering the loop 4 hot leg. Following a loss-of-coolant accident, direct hot leg injection would be used to prevent undesirable precipitation of boron in the core from blocking coolant flow. Charging water can also be directed into any loop from the low pressure safety injection header, which is normally isolated from the charging and volume control system.

Dilution of boron concentration in the main coolant system or the low pressure surge tank can be terminated by (1) isolation of the charging system via motor-operated valves, (2) shutting off charging pumps or the low pressure surge tank make-up pump, or (3) by isolation of the primary make-up water system. The low pressure surge tank level indication is equipped with high and low alarms that could provide indications of a possible change in boron concentration.

An inadvertent boron dilution would require multiple operator errors and procedural violations or multiple equipment malfunctions. This could lead to an increasing core reactivity level either from subcritical conditions or while operating at power. Analysis of this event has been performed for each mode of plant operation permitted by the Technical Specifications. An evaluation was also performed of the consequence of a failure to borate during a controlled cooldown, which could also result in a reactivity increase. The analysis reported below was submitted to the NRC via Reference VI.1, the Core XV performance analysis.

VI.B Plant Protection

An automatic reactor trip would terminate any reactivity transient due to boron dilution or failure to borate. The available trip functions are (1) high neutron flux, (2) high startup rate, (3) high pressurizer water level, and (4) high main coolant loop pressure (effective with Core XV operation). In addition, a manual scram may be inserted by the operator based upon many indications and alarms. Standard Review Plan 15.4.6 presents minimum allowable time intervals for operator recognition and manual action to avoid complete loss of shutdown margin. During startup, cold shutdown, hot standby, and power operation, an interval of 15 minutes is specified. During refueling, the interval specified for recognition and prevention is 30 minutes. Boron dilution at power operation is less limiting than an inadvertent rod-withdrawal accident (Topic XV-8), for which automatic protection is adequate.

VI.C Analysis

VI.C.1 Boron Dilution During Mode 1 (Power Operation) and Mode 2 (Startup)

The Technical Specifications currently require that all four primary loops must be operating in these modes. However, one loop is assumed to be isolated to introduce conservatism in the analysis. This assumption reduces the available volume for boron dilution from 2900 ft^3 to 2400 ft^3, resulting in a higher rate of reactivity change. The maximum capacity of all three charging pumps is assumed, 100 gpm, although it is unlikely that all three would be used to deliver makeup water to the primary system. The minimum initial shutdown margin is 4.72% , which is considerably less than normally exists during operation in Modes 1 and 2. Minimum inverse boron worth was also assumed. Using these assumptions, the minimum time available for operator recognition and action exceeds 45 minutes.

Should the reactor be in the automatic control mode during a boron dilution at power, the control rod group selected as the controlling group by the operator would insert to offset any resulting temperature change. Even if rods move at the rate of 6 in/min, it would require 15 minutes to fully insert the control group. During this time, the operator would be alerted by the high average temperature and high core power alarms. In addition, the vapor container intercom system would normally provide an audible indication of rod motion. These indications and alarms would alert the operator in sufficient time to terminate the dilution and restore adequate shutdown margin.

VI.C.2 Boron Dilution During Mode 3 (Hot Standby)

At least one loop must be in operation during Mode 3, with a minimum of 1% $\Delta\rho$ subcriticality and >4.72% $\Delta\rho$ shutdown margin. The maximum charging flowrate of 100 gpm is assumed, although feed and bleed operations are not normally conducted using all charging pumps. If this occurred, four alarms would indicate to the operator that boron dilution could be occurring. These alarms are (1) high flux recorder alarm, (2) pressurizer high level alarm, (3) pressurizer high pressure alarm, and (4) pressurizer surge line low temperature alarm. In addition the bleed line radiation alarm could occur.

Results show that loss of shutdown margin during a boron dilution event in Mode 3 would not occur for at least 30 minutes.

VI.C.3 Boron Dilution During Mode 4 (Hot Shutdown), Mode 5 (Cold Shutdown), and Mode 6 (Refueling)

The shutdown cooling system may be used during operations in either Mode 4, Mode 5, or Mode 6. Thus, the active main coolant volume excluding could be approximately 1486 ft^3. Assuming that the upper vessel head is drained, the minimum volume during Mode 6 operation (refueling) could be 1276 ft^3. This volume was assumed for conservatism to apply to Mode 4 and Mode 5 operation. A Technical Specification requiring 4% $\Delta\rho$ subcriticality and 5% $\Delta\rho$ shutdown margin when in Modes 4 and 5 was proposed in Reference IV.1.

This allows partial withdrawal of a control rod group to provide 1% $\Delta\rho$ reactivity worth, which is required by Technical Specification. Control rods are not routinely withdrawn during the Mode 4 or Mode 5 operation, however, a requirement of 4% $\Delta\rho$ subcriticality and 5% $\Delta\rho$ shutdown margin will allow 1% $\Delta\rho$ to be withdrawn.

Assuming 100 gpm charging flow and the most limiting combination of coolant temperature, initial boron concentration, and inverse boron worth, more than 20 minutes is available for operator recognition and action before shutdown margin is lost. This time period is sufficient for operator acknowledgement and corrective action following alarms of high flux level, or Low Pressure Surge Tank high level, high temperature, or high pressure.

VI.C.4 Failure to Borate Prior to Cooldown

Because of the large negative temperature coefficient of reactivity at end of cycle, any decrease in main coolant system temperature increases the core reactivity state. Consequently, during the process of cooldown of the main coolant system, adequate shutdown margin must exist from boron concentration and control rod worth.

Failure to ensure adequate shutdown margin prior to cooldown was evaluated using the following basic assumptions:

a) The moderator temperature coefficient is the most negative value expected with all rods in the core, including uncertainties.

b) The reactor is initially 1% subcritical at an average temperature of 515°F, the maximum allowed temperature at hot standby (Mode 3) conditions.

c) The shutdown margin at 515oF is 4.72%$\Delta\rho$, less than the minimum required by the Technical Specifications.

d) The main coolant system temperature is reduced at the maximum allowable rate of 50oF/hr.

In order to make the reactor critical from these initial conditions, the average coolant temperature must be reduced to approximately 490oF. This temperature reduction requires approximately 30 minutes, which is ample time for the operator to diagnose the condition and take necessary corrective action. A cooldown below 350oF would be required for a complete loss of shutdown margin. This would require in excess of 3 hours, which is adequate time for the operator to correct the situation.

VI.C.5 Conclusion

The probability of erroneous dilution is considered very small because of the equipment, controls, and administrative procedures provided for boron dilution activities. However, in the unlikely event that an unintentional dilution of boron in the main coolant system occurs, numerous alarms and indications are available to alert the operator of the condition. If the reactor is critical at the time dilution begins, automatic safety features of the reactor protection system would ensure acceptable plant performance without operator intervention. For boron dilutions initiated during any operational mode, adequate time exists for the operator to determine the cause of the dilution and take corrective action before a complete loss of shutdown margin occurs.

VI.D References

VI.1 FYR81-52, YAEC Letter to NRC, Core XV Refueling Proposed Change No. 173, 26 March 1981.

VII. Topic XV-12: Spectrum of Rod Ejection Accidents (PWR)

The mechanism postulated to result in ejection of a control rod from the core is a mechanical failure of the control drive mechanism housing (CRDM). The CRDM housing and CRDM nozzle are an extension of the reactor coolant system boundary and are designed and manufactured to Section VIII of the ASME Boiler and Pressure Vessel Code (1956 Edition). Rod ejections are considered to be a limiting-fault event by Reference VII.1 ANSI N18.2 standards, which is an event that is not expected to occur within the lifetime of the plant. It is postulated for analysis because of the potential for radiation release due to possible fuel pin damage caused by extreme temperatures. NRC Regulatory Guide 1.77, Reference VII.2, places a limit of 280 cal/gm on the maximum radially-averaged fuel enthalpy during the accident. If the radially-averaged enthalpy the fuel pin hot spot exceeds 280 cal/gm, by NRC standards the pin is assumed to fail and all gap radioactivity is assumed to be released. Currently, however, Yankee Atomic Electric Company employs more conservative criteria to measure fuel damage. These criteria are identical to those adopted by Combustion Engineering for preparation of the Maine Yankee Plant Final Safety Analysis Report. They are as follows:

Failure Mode	Basis	Fuel Enthalpy Criteria
Clad Damage	Radial Average at Axial Hot Spot	≥200 cal/gm
Incipient Fuel Melting	Axial Hot Spot	≥250 cal/gm
Fully Molten Fuel	Axial Hot Spot	≥310 cal/gm

The amount of reactivity which can be introduced into the core as a result of a rod ejection is minimized by operation with chemical reactivity shim. Most of the control rods are withdrawn before reaching full power, thus minimizing the severity of any rod ejection accident. Furthermore, should a rod ejection occur, the resulting high power level initiates a high neutron flux trip signal which causes the shutdown rods to insert, thus reducing the neutron generated power to negligible levels. The loss of coolant resulting from the primary system rupture and its consequences are similar to those for small breaks, which are discussed in the section describing the Loss-of-Coolant Accident, Topic VI-2.D.

VII.B Analysis

The analysis of the rod ejection accident is performed using the CHIC-KIN digital computer program to simulate the core hot channel response. This model incorporates the standard six-delayed neutron groups representation along with explicit reactivity contributions from rod motion, fuel temperature effect and moderator temperature variations.

The principal reactivity feedback mechanism affecting the course of the nuclear transient is caused by fuel temperature increase. Although the axial shape does not change significantly during the course of a rod ejection accident, the radial shapes at various axial slices undergo marked changes. However, the use of the static, non-Doppler flattened radial pin peaking factor (obtained from two-dimensional diffusion theory results) in conjunction with the average core energy release (obtained from the point kinetics results) yields hot spot energy releases that are conservative, since the fuel temperature effect during a transient is strongest at the radial peak location. Thus, the radial peak is limited to a value below that obtained from the static ejected rod configuration for the major portion of the transient.

Rod ejections are evaluated for each reload core using methods similar to the Core XI analysis, submitted via Reference VII.3 and approved by the NRC via Reference VII.4. Typically, results vary from core to core because of changes in rod worth, moderator and fuel temperature reactivity coefficients, axial power distributions, and radial two-dimensional peaking factors. Analysis is performed for both zero power and full power operating conditions.

Most recently, the rod ejection analysis for Core XV, Reference VII.5, showed results to be less limiting than for Core XIV. Core XI results were more limiting than either Core XIV or Core XV. The following assumptions applied for the Core XV analysis:

1. The most worthy control rod ejects instantaneously;
2. The fuel temperature coefficient is the least negative value expected throughout core life during power operation;
3. The ejected rod worth is the maximum throughout core life;
4. The calculated maximum ejected rod peaking throughout core life is increased by 10 percent in order to account for calculational uncertainties.

Results for Core XV were that radially-averaged enthalpy for both low power and full power cases was less than 200 cal/gm, so no fuel pin damage was predicted.

In conclusion, together with established rod insertion limits, the reactor protection system ensures that no clad damage occurs during a rod ejection transient.

VII.C References

VII.1 ANSI N18.2, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, 1973.

VII.2 Regulatory Guide 1.77, Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors, US AEC, May 1974.

VII.3 YAEC Letter to NRC, Proposed Change No. 115 (Core XI Refueling), 29 March 1974.

VII.4 DOL/AEC Letter to YAEC, Amendment No. 9, 30 July 1974.

VII.5 FYR81-52, YAEC Letter to NRC, Core XV Refueling, 26 March 1982.

YANKEE NUCLEAR POWER STATION

Topic XV-14 - Inadvertent Operation of ECCS or CVCS Malfunction that causes an increase in reactor coolant inventory.

An increase in main coolant system inventory can result from inadvertent safety injection or from a malfunction in the pressurizer level control system (CVCS). Depending upon boron concentration and temperature of the injected water, a power level increase could result.

A malfunction of the pressurizer level control system could at most result in the starting of all three charging pumps and in the closing of all three letdown orifices. This situation results in a maximum charging rate of 100 gpm (33 gpm/pump). At this rate, the pressurizer level will increase at approximately 17 inches per minute from its normal level of 120 inches (1/3 full). A high pressurizer water level trip will scram the reactor when the pressurizer level reaches 200 inches. However, a high main coolant system pressure trip would most likely have already occurred when the MCS pressure reached 2200 psig. During the level increase, a high pressurizer level alarm will occur at 150 inches, alerting the operator to the overcharging condition, giving him time to stop the charging pumps before the MCS reaches an overpressure condition.

In addition to the pressurizer level and MCS pressure increase, the overcharging condition could result in a boron dilution of the MCS resulting in an increase in reactor power. Should the reactor be in the automatic control mode during a boron dilution at power, the control rod group selected by the operator would insert to offset any temperature increase resulting from the core power-steam flow mismatch. During this time, the operator would be alerted by the high average temperature and high core power alarms. In addition, the Vapor Container Intercom System would normally provide an audible indication of rod motion. These indications and alarms would alert the operator of the condition in sufffcent time to terminate the dilution and restore adequate shutdown margin. If the reactor should be in the manual mode, the same alarms and indications would occur but the control rods would not be inserted. Boron dilution at power is discussed in detail in the Core XV licensing submittal, Reference XV-14A.

Inadvertent operation of the ECCS during normal operating conditions will not result in adverse consequences since no flow will be delivered to the MCS due to the fact that the ECCS shutoff head of 1560 psig which is well below MCS operating pressure. Individual recirculation lines from each HPSI pump discharge to the Safety Injection tank (SIT) provide minimum recirculation flow pump protection. In addition, inadvertent HPSI pump operation will not result in overpressurization of the LPSI pumps. First, it is highly improbable that the five check valves in series would leak excessively. Second, the respective heads of the LPSI pumps and HPSI pumps are of nearly the same magnitude - 870 psi for the HPSI pumps and 690 psi for the LPSI thus damage to either pump is unlikely. Third, flow back to the SIT through LPSI recirculation orifices would prevent pressure buildup.

The inadvertent operation of the ECCS during low temperature/low pressure conditions was addressed in the evaluation of the reactor Low Temperature Overpressure Protection System (LTOP). Pressure transients resulting from the mass addition from one SIS train between MCS temperatures of 300°F to 324°F and from the mass addition from one low pressure safety injection pump below 200°F MCS temperature were excluded as design considerations from the Yankee Rowe LTOP system. The justification for excluding these events is best summarized in the NRC's Safety Evalution of the Yankee Rowe Low Temperature Overpresure Protection System, Reference XV-14B.

Based upon this review, it is concluded that the Yankee Nuclear Power Station is adequately designed to cope with an inadvertent operation of the ECCS or CVCS.

References XV-14

XV-14A "Yankee Nuclear Power Station Core XV Performance Analysis", YAEC-1240, March 1981.

XV-14B "Safety Evaluation by the Office of Nucear Reactor Regulation Supportiong Amendment No. 59, Facility Operating Licesnse No. DPR-3, Yankee Atomic Electric Company, Yankee Nuclear Power Station (Yankee Rowe), Docket No. 50-29", USNRC, September 14, 1979.

YANKEE NUCLEAR POWER STATION

Topic XV-15 - Inadvertent opening of a PWR pressurizer safety/relief valve

The design pressure of the Yankee Rowe main coolant system is 2485 PSIG. Normal system operating pressure is 2,000 PSIG. Two pressurizer code safety valves and a single pressurizer power operated relief valve are provided to limit any main coolant pressure transient to 110% of the design pressure (2735 PSIG). In addition, a 1-inch water relief valve (90 GPM at 2735 PSIG) is provided in each main coolant loop to prevent overpressurization of an isolated main coolant loop. Technical specifications do not allow power operation with an isolated loop.

The two pressurizer code safety valves are designed to relieve 92,000 lbs per hour of saturated steam at set pressures of 2485 PSIG and 2560 PSIG respectively. The combined relief capacity of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first RPS setpoint is reached and no credit for PORV or steam dump valve operation. The single power operated pressurizer relief valve has a capacity of 70,000 lbm per hour at its set pressure of 2385 PSIG. It is provided primarily to limit challanges to the code safety valves but in certain accident situations it may be relied upon to depressurize the MCS and provide for core cooling via feed and bleed. It should also be noted that PORV actuation would not prevent a reactor trip in an overpressurization event since its set pressure is well above the high main coolant system pressure trip of 2,200 PSIG

Since the normal system pressure at Yankee is 2,000 PSIG, considerable margin exists between that pressure and the set pressures of the PORV (2385 PSIG) and the code safety valves (2485 psia and 2560 psia). Thus, the most likely time for an inadvertent relief or safety valve blowdown to occur would be following an event such as a loss of load without prompt reactor trip in which a relief or safety valve lifts and fails to reseat. In the event of the PORV remaining open, the operator can terminate the ensuing blowdown by closing the PORV block valve. Since the safety valves cannot be isolated, blowdown from a stuck-open safety valve would continue until either the valve reseats or until the plant is brought to cold shutdown.

It should be noted that in 20 years of Yankee operation, neither the code safety valves nor the PORV have been challenged. This is due to the wide margin between operating pressure and valve setpoints as noted above. With the installation of the new MCS high pressure trip of 2,200 PSIG for Core XV, the potential for future challenges to the PORV or the code safety valves is substantially reduced from the already low probability that currently exists.

In the unlikely event of a stuck-open pressurizer PORV or safety valve, blowdown from the main coolant system via the pressurizer to containment would occur. A 240 PSIG rupture disk is provided on the relief valve discharge piping. Perforation of the rupture disk would direct the blowdown into containment. Relief valve discharge piping is designed such that normal valve "weeping" is directed to the low pressure surge tank and any other higher flow rate would cause the ruptue disk to blow to containment.

The rate of MCS depressurization from the stuck-open valve would depend upon the venting capacity of the relief or safety valve in the unseated condition. Reactor scram would occur upon reaching the low main coolant system pressure setpoint of 1,800 PSIG. The rate of depressurization following reactor trip increases because of the cooldown of the primary system. The ensuing system "shrink" results in a rapid transition from 1,800 PSIG MCS pressure to approximately 935 PSIG or 760 PSIG depending upon the secondary side configuration.

The automatic rod control system at Yankee would not attempt to maintain constant reactor power prior to reactor trip since it has a rods in function only which is controlled by the average coolant temperature ($T_{average}$). As such, the automatic rod control system can only serve to decrease reactor power in such an event.

A safety injection actuation signal (SIAS) will occur when the pressure drops below 1,700 PSIG, thus starting safety injection pumps. The high pressure and low pressure safety injection pumps start on SIAS but do not begin delivering flow to the MCS until the MCS pressure drops below their combined shut off head of 1,560 PSIG. This will occur quite quickly following reactor trip. Once the MCS temperature equilibrates with the secondary side temperature, MCS depressurization continues (at a slower rate due to saturated conditions) until the ECCS recovers the situation, refills the MCS and long term cooling is established.

The inadvertent opening of a pressurizer relief or pressurizer safety valve has not been specifically analyzed. Such an event would represent a break area equivalent to an internal diameter of 0.8 inches for the PORV to approximately 0.9 inches for the safety valves. Since these breaks are on the hot side of the system, the system effects are less severe than for the equivalent size cold leg break. The system response to inadvertent opening of the pressurizer relief or safety valves are bounded by the small break LOCA analyses performed for Core XIII which meet Appendix K criteria (References XV-15A and XV-15B). That analysis consisted of a spectrum of cold leg breaks with ECCS spillage ranging in size from 2.25 inches ID to 10.0 inches ID. The limiting break was identified to be a 4.0 inch ID break. Analysis of any breaks less than 4 inches ID would prove to be non-limiting.

The system response to a stuck open PORV or safety valve is much less limiting than the equivalent cold leg break for several reasons. First, the break is at the highest point in the MCS, facilitating steam venting and ultimately system refill. Second, safety injection will occur at all four loop injection points versus only three in the Core XIII Analysis since no pipe break exists, maximizing system refill. Finally, the relieving capacity of these valves are such that decay heat can be removed from any of them early in the blowdown transient. For example, the PORV can remove decay heat by removing steam at approximately 15 minutes to one-half hour into the accident. This capability allows the initial inventory on the secondary side of the steam generators to handle decay heat requirements amply until the PORV handles it on its own coupled with ECCS injection.

Based upon this review, it is concluded that the Yankee Nuclear Power Station is adequately designed to cope with an inadvertent opening of a pressurizer relief or safety valve.

References

XV-15A Proposed Change No. 145, Supplement No. 5, Yankee Rowe Core XIII ECCS Performance Evaluation, August 1, 1977.

XV-15B Proposed Change No. 145, Supplement No. 7, WYR 77-90, Additional Yankee Rowe Core XIII Small Break Analysis, September 21, 1977.